May 4, 2016

Mr. Edward D. Halpin
Senior Vice President and
Chief Nuclear Officer
Pacific Gas and Electric Company
P.O. Box 56
Mail Code 104/6
Avila Beach, CA 93424

SUBJECT: HUMBOLDT BAY POWER PLANT UNIT 3 - ISSUANCE OF AMENDMENT RE: LICENSE TERMINATION PLAN (TAC NO. J00485)

Dear Mr. Halpin:

The Commission has issued the enclosed Amendment No. 45 to Facility Operating License No. DPR-7 for the Humboldt Bay Power Plant Unit 3. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated May 3, 2013, as supplemented on February 14, 2014, April 2, 2014, May 13, 2014, August 13, 2014, and March 16, 2015.

The amendment revises the Humboldt Bay Power Plant, Unit 3 License to add License Condition 2.C.(5) to the Humboldt Bay license. This new license condition incorporates the NRC approved "License Termination Plan" (LTP), and associated addendum, into the Humboldt Bay license and specifies limits on the changes the licensee is allowed to make to the approved LTP without prior NRC review and approval.

A copy of our related Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

***/RA/***

John B. Hickman, Project Manager
Reactor Decommissioning Branch
Division of Decommissioning, Uranium Recovery
and Waste Programs
Office of Nuclear Material Safety and Safeguards

Docket No. 50-133

Enclosures:
1. Amendment
2. Safety Evaluation

cc w/enclosures:
Humboldt Bay Service List

May 4, 2016

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Chief Nuclear Officer
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Humboldt Bay Service List

DISTRIBUTION: DCD r/f OGC ACRS/ACNW RKeller, RIV

**ML15090A444**

| OFFICE | DUWP:PM | DUWP:LA | DUWP:BC | OGC | DUWP:D | DUWP:PM |
|---|---|---|---|---|---|---|
| NAME | JHickman | CHolston | BWatson | APessin | JTappert | JHickman |
| DATE | 3 / 31 / 2015 | 4 / 1 / 2015 | 6 / 9 / 2015 | 12/18/2015 | 5/4/2016 | 5/4/2016 |

**OFFICIAL RECORD COPY**

Humboldt Bay Power Plant, Unit 3 Service List

cc:

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Senior Vice President and
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<u>PACIFIC GAS AND ELECTRIC COMPANY</u>

<u>DOCKET NO. 50-133</u>

<u>HUMBOLDT BAY POWER PLANT, UNIT 3</u>

<u>AMENDMENT TO FACILITY OPERATING LICENSE</u>

Amendment No. 45
License No. DPR-7

1. The Nuclear Regulatory Commission (the Commission or NRC) has found that:

   A. The application for amendment by Pacific Gas and Electric Company (the licensee), dated May 3, 2013, as supplemented February 14, 2014, April 2, 2014, May 13, 2014, August 13, 2014, and March16, 2015, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

   B. The facility will be maintained in conformity with the application, as amended, the provisions of the Act, and the applicable rules and regulations of the Commission;

   C. There is reasonable assurance: 1) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public; and 2) that such activities will be conducted in compliance with applicable portions of the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended as follows:

License Condition 2.C.(5) is added to read as follows:

5. <u>License Termination Plan (LTP)</u>

NRC License Amendment No. 45 approves the LTP. In addition to the criteria specified in 10 CFR 50.59 and 10 CFR 50.82(a)(6), a change to the LTP requires prior NRC approval if the change:

(a) Increases the probability of making a Type 1 decision error, as that term is described in the NRC's Multi-Agency Radiation Survey and Site Investigation Manual, NUREG-1575, Revision 1 (August 2000) (MARSSIM), above the level stated in the LTP;

(b) Increases the radionuclide-specific derived concentration guideline levels (DCGL), as that term is described in the MARSSIM, and related minimum detectable concentrations;

(c) Increases the radioactivity level, relative to the applicable DCGL, at which investigation occurs;

(d) Changes the statistical test applied other than the Sign Test or Wilcoxon Rank Sum Test;

(e) Results in significant environmental impacts not previously reviewed.

Reclassification of survey areas, as described in MARSSIM, from a less to a more restrictive classification (e.g., from a Class 3 to a Class 2 area) may be done without prior NRC notification; however, reclassification to a less restrictive classification (Class 1 to Class 2 area) will require NRC notification at least 14 days prior to implementation.

3. This license amendment is effective as of the date of issuance and shall be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

***/RA/***

John R. Tappert, Director
Division of Decommissioning, Uranium Recovery
and Waste Programs
Office of Nuclear Material Safety and Safeguards

Date of Issuance: May 4, 2016

ATTACHMENT TO LICENSE AMENDMENT NO. 45

FACILITY OPERATING LICENSE (POSSESSION ONLY) NO. DPR-7

DOCKET NO. 50-133

Replace the following pages of the License with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

| REMOVE | INSERT |
|---|---|
| 4 | 4 |

5. License Termination Plan (LTP)

NRC License Amendment No. 45 approves the LTP. In addition to the criteria specified in 10 CFR 50.59 and 10 CFR 50.82(a)(6), a change to the LTP requires prior NRC approval if the change:

(a) Increases the probability of making a Type 1 decision error, as that term is described in the NRC's Multi-Agency Radiation Survey and Site Investigation Manual, NUREG-1575, Revision 1 (August 2000) (MARSSIM), above the level stated in the LTP;

(b) Increases the radionuclide-specific derived concentration guideline levels (DCGL), as that term is described in the MARSSIM, and related minimum detectable concentrations;

(c) Increases the radioactivity level, relative to the applicable DCGL, at which investigation occurs;

(d) Changes the statistical test applied other than the Sign Test or Wilcoxon Rank Sum Test;

(e) Results in significant environmental impacts not previously reviewed.

Reclassification of survey areas, as described in MARSSIM, from a less to a more restrictive classification (e.g., from a Class 3 to a Class 2 area) may be done without prior NRC notification; however, reclassification to a less restrictive classification (Class 1 to Class 2 area) will require NRC notification at least 14 days prior to implementation.

D. This license amendment is effective as of the date of issuance and shall expire at midnight, November 9, 2015

FOR THE NUCLEAR REGULATORY COMMISSION

Lester S. Rubenstein, Acting Director
Standardization and Non-Power
Reactor Project Directorate
Division of Reactor Projects III, IV,
V and Special Projects
Office of Nuclear Reactor Regulation

Enclosure:
Appendix A – Technical
Specifications

Date of Issuance: July 19, 1988

SAFETY EVALUATION BY OFFICE OF NUCLEAR MATERIAL SAFETY AND SAFEGUARDS

RELATED TO AMENDMENT NO. 45 TO FACILITY OPERATING LICENSE NO. DPR-7

PACIFIC GAS AND ELECTRIC COMPANY

HUMBOLDT BAY POWER PLANT, UNIT 3

DOCKET NO. 50-133

## 1.0 INTRODUCTION

On May 3, 2013, Pacific Gas and Electric Company (PG&E, the licensee)[1] submitted a License Amendment Request (LAR), "Addition of License Condition 2.C.5, 'License Termination Plan,'" for Humboldt Bay Power Plant Unit 3 (HBPP or the facility) (Agencywide Documents Access and Management System Accession number ML13130A008), for NRC review and approval. On March 31, 2014 (ML14093A050), PG&E submitted additional information, including site-specific decommissioning cost information, as specified in the LAR. Subsequently, on August 13, 2014 (ML14246A164), the licensee submitted Revision 1 to the LTP which reflects, among other things, requests for additional information associated with Revision 0. According to the licensee, the LTP demonstrates that the remaining decommissioning activities will be performed in accordance with 10 CFR Part 50, will not be adverse to the common defense and security or to the health and safety of the public, and will not have a significant adverse effect on the quality of the environment.

The approval of the LTP will allow the licensee to take the final steps that will result in the termination of its 10 CFR Part 50 operating license. In accordance with the requirements of section 50.82(a)(9) of Title 10, U.S. Code of Federal Regulations (10 CFR 50.82(a)(9)), the licensee submitted a license termination plan for its facility. Section 50.82(a)(9) states in part: "All power reactor licensees must submit an application for termination of license. The application for termination of license must be accompanied or preceded by a license termination plan to be submitted for NRC approval." Under the provisions of 10 CFR 50.82(a)(10), the U.S. Nuclear Regulatory Commission (NRC) approves license termination plans by license amendment. Thus, the licensee has requested the addition of a new License Condition to the HBPP License. The new license condition would incorporate the NRC approved License Termination Plan (LTP) into the HBPP license, and allow the licensee to make certain changes to this approved LTP without prior NRC review or approval. The new License Condition read as follows:

> 2.C.5 License Termination Plan (LTP)
>
> NRC License Amendment No. 45 approves the LTP. In addition to the criteria specified in 10 CFR 50.59 and 10 CFR 50.82(a)(6), a change to the LTP requires prior NRC approval if the change:

---

[1] The terms "PG&E" and "licensee" are used interchangeably throughout this document.

(a) Increases the probability of making a Type I decision error above the level stated in the LTP;

(b) Increases the radionuclide-specific derived concentration guideline levels (DCGL) and related minimum detectable concentrations;

(c) Increases the radioactivity level, relative to the applicable DCGL, at which investigation occurs;

(d) Changes the statistical test applied other than the Sign Test or Wilcoxon Rank Sum Test;

(e) Results in significant environmental impacts not previously reviewed.

Reclassification of survey areas from a less to a more restrictive classification (e.g., from a Class 3 to a Class 2 area) may be done without prior NRC notification; however, reclassification to a less restrictive classification (e.g., Class 1 to Class 2 area) will require NRC notification at least 14 days prior to implementation.

The staff sent a Request for Additional Information (RAI) to the licensee on December 24, 2013. The licensee responded by letters dated February 14, 2014, April 2, 2014, and May 13, 2014. The licensee provided clarifying information by e-mail dated March 16, 2015.

## 2.0 BACKGROUND

Humboldt Bay Power Plant (HBPP) Unit 3 was issued an operating license on August 28, 1962. On July 2, 1976, HBPP Unit 3 was shut down for annual refueling and to conduct seismic modifications. In 1983, updated economic analyses indicated that restarting HBPP Unit 3 would probably not be cost-effective, and in June 1983, Pacific Gas and Electric Company (PG&E), the HBPP licensee, announced its intention to decommission the unit. On July 16, 1985, the NRC issued Amendment No. 19 to the HBPP Unit 3 Operating License (Reference 1) to change the status to possess-but-not-operate, and the plant was placed into a SAFSTOR[2] status.

On December 15, 2003, PG&E submitted a license application (Reference 2) to the NRC in accordance with Part 72 of Title 10 of the *Code of Federal Regulations* (10 CFR) to construct and operate an Independent Spent Fuel Storage Installation (ISFSI) on the HBPP site. On November 17, 2005, the NRC issued site-specific Materials License No. SNM-2514 to PG&E for an ISFSI at the HBPP site (Reference 3.) The transfer of spent fuel from the fuel storage pool to the ISFSI was completed in December 2008, and the decontamination and dismantlement phase of HBPP Unit 3 decommissioning is actively underway.

## 3.0 TECHNICAL EVALUATION

The licensee submitted its LTP in accordance with 10 CFR 50.82(a)(9), which requires the LTP to contain the following information: (1) a site characterization; (2) identification of remaining dismantlement activities; (3) plans for radiological site remediation; (4) detailed plans for the conduct of final radiation survey; (5) a description of the end use of the site, if a restricted option

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[2] SAFSTOR is an NRC approved method of decommissioning a nuclear facility where the nuclear facility is placed and maintained in such condition that the nuclear facility can be safely stored and subsequently decontaminated to safe levels.

is selected; (6) an updated site-specific estimate of remaining decommissioning costs; and (7) a supplement to the environmental report, pursuant to 10 CFR 51.53, describing any new information or significant environmental changes associated with the licensee's proposed termination activities. In addition, the licensee requested the authority to make certain changes to the LTP, once approved by NRC. This proposed change authority is delineated in the proposed license condition. The LTP describes HBPP's approach for demonstrating compliance with radiological criteria, for unrestricted use. That criteria is set forth in 10 CFR 20.1402, namely, that the annual dose limit is 0.25 mSv (25 mrem) total effective dose equivalent (TEDE) above background from all pathways, including groundwater. Section 20.1402 also requires that the residual radioactivity be reduced to levels that are "as low as is reasonably achievable" (ALARA).

## 3.1 Site Characterization

### 3.1.1 Facility Radiological Status

Licensees conduct site characterization surveys to determine the nature and extent of radioactive contamination in buildings, plant systems and components, site grounds, and surface and groundwater. The major objectives of characterization activities are to: permit the planning and conduct of radiological remediation activities; confirm the effectiveness of radiological remediation methods; provide information to develop specifications for final status surveys (FSS); define site and building areas as survey units and assign survey unit classifications; and provide information for dose modeling. HBPP conducted radiological site characterization activities that included a historical site assessment[3] (HSA) (ML13130A135 & ML13130A134), scoping surveys, and a characterization survey. HBPP conducted an HSA that included a review of records maintained to satisfy the requirements of 10 CFR 50.75(g)(1), as well as environmental reports, Radiological Environmental Monitoring Reports, Radioactive Effluent Release Reports, Licensee Event Reports, Plant Operating Reports, Plant Safety Analyses, Radiological Surveys, and Plant Operating Logs (PG&E, 2014). Additionally, HBPP conducted personal interviews of current and former HBPP site personnel and reviewed collections of photographs to verify and validate the HSA.

The licensee performed personnel interviews during the site inspection and by telephone while conducting the HSA. The licensee selected HBPP maintenance and radiation protection workers employed during the time that Unit 3 was in operation at HBPP. The licensee did not discover any undocumented events during this process and found that the interviews proved beneficial in evaluating the past radiological operations at HBPP. HBPP included a selection of photographs in the LTP to provide a visual assessment of the HSA. The licensee did not conduct a site reconnaissance consistent with guidance in Section 3.5 of the NRC's Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM), NUREG-1575 (NRC, 2000). The licensee based its justification of not performing a site reconnaissance because the licensee had continuous occupancy of the HBPP site and has maintained detailed information in HBPP's records and other documents. Additionally, the licensee conducted personnel interviews to verify locations and current conditions of questionable items or issues, such as radioactive liquid spills or the spread of radiological contamination that the licensee discovered during these investigations.

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[3] In this context, the term "historical site assessment" concerns the assessment of radiological contamination on the site (essentially, a history of the radioactivity on the site as a result of the HBPP Unit 3 operation). The historical site assessment does not concern historic properties or cultural resources.

LTP Section 2.1.5 provides a description of the operational history of the HBPP beginning with the construction and provisional operating permits the Atomic Energy Commission (AEC) issued in October 1960 and August 1962, respectively, to the unit shutdown in 1976, and subsequent decommissioning activities to the present. LTP Table 2-1 summarizes the operational and post-operational history. The licensee describes the radiological waste handling procedures, criteria used to determine radiological soil contamination, initial soil segregation and radiological decontamination procedures, major site dismantlement activities within and outside of the restricted area (RA), and the radiological sources within and outside of the RA. The licensee removed spent fuel and shipped it to a reprocessing facility during plant operation, with the last shipment in 1971. HBPP removed all the stainless steel clad fuel assemblies that were prone to failure during this phase. Spent fuel remained onsite in the spent fuel pool (SFP) until December 2008, when the licensee transferred the fuel to the independent spent fuel storage installation (ISFSI), that is separately licensed by the NRC under 10 CFR Part 72.

HBPP excavated soils while constructing buildings and roads during and after reactor operations at the site. The licensee surveyed and analyzed soils using a lower limit of detection of 0.18 pCi/g Cs-137. The licensee considered soil contaminated if samples contained fission or activation produced isotopes related to HBPP operations other than Cs-137. HBPP also considered soils contaminated if the licensee measured Cs-137 greater than 6 inches from the surface or less than 6 inches from the surface, but containing Cs-137 concentrations greater than 0.4 pCi/g. HBPP disposed contaminated soils at an NRC-licensed disposal site and placed non-contaminated soils either west or east of the discharge canal.

The licensee conducted radiological site characterizations in 1997 and 2008. Within the RA, the licensee has removed the Unit 3 turbine and generator, the reactor vessel head, the reactor feed pumps, the dry well shield plug, the SFP storage racks, the reactor vessel internals, and is segmenting the reactor vessel. Outside of the RA, the licensee has removed the Fossil units 1 and 2, the fuel oil tanks 1 and 2, Unit 1 and 2 transformers, and is installing the slurry wall. The licensee has temporary power supplying HBPP Unit 3 during the decommissioning process. The licensee has identified all areas within the RA affected radiologically by the operation of the facility, unplanned events, or subsequent decommissioning activities. The licensee identified contaminated areas outside of the RA have been affected by radiological events, and deposition of effluent radiological releases (e.g., stack and/or liquid) releases. The licensee states that it has not identified radiological contamination in the Humboldt Bay area; however, the Humboldt Bay area will be classified pending further characterization. LTP Table 2-2 details events and/or issues, such as spills and steam condensate leaks, that affected the site, and Figure 2-1 illustrates the locations of these events described. Section 2.1.7 describes the survey unit identification and classifications that the licensee used and provides a summary of this information in Table 2-3. Figure 2-2 identifies the radiological impacted and non-impacted areas, and illustrates the proposed survey locations. LTP Table 2-3 clearly associates the survey area designator to the building or site location and annotates whether or not the area is within or outside the RA, describes the total area of the survey area, and describes the classification (e.g., Class 1, 2, or 3).

To make the best use of resources for decommissioning, MARSSIM places greater survey efforts on areas that have, or had, the highest potential for contamination. Areas that have no reasonable potential for residual contamination are classified as non-impacted areas. Class 1 Areas have, or had prior to remediation, a potential for radioactive contamination (based on site operating history) or known contamination (based on previous radiation surveys) above the DCGLw. Class 2 Areas have, or had prior to remediation, a potential for radioactive contamination or known contamination, but are not expected to exceed the DCGLw. Class 3

areas are not expected to contain any residual radioactivity, or are expected to contain levels of residual radioactivity at a small fraction of the DCGLw, based on site operating history and previous radiation surveys.

Section 2.1.8 provides a summary of the area radiological impact. The licensee conducted the initial radiological site characterization in 1997 and selected the following isotopes for analyses from past sampling documentation: Fe 55, Ni 63, Mn 54, Co 60, Sr 90, Tc 99, Am 241, Pu 238, Pu 239/240, Cm 242, Cm 244, Zn 65, Sb 125, Cs 134, Cs 137, Eu 154, Eu 155, and Pu 241. The licensee analyzed all samples for gamma emitters and randomly selected and analyzed samples for hard-to-detect (HTD) isotopes, such as Ni-63 and Fe-55. The licensee reported that samples contained very low soil concentrations of Am-241 and Ni-63 relative to the soil concentrations of Cs-137. The licensee has identified seven Class 1 sites to include the proposed future waste management building (WMB), five Class 2 sites, and the remaining areas Class 3. Tables 2-4 through 2-19 provide the characterization data collected in these areas.

The six present Class 1 sites are the open land area inside the RA north of Unit 3 designated as NOL01; the southern section of the discharge canal (OOL01- Discharge Canal South); the open land within the eastern portion of the intake canal (OOL02-Intake East); the open land outside of the RA and north of Units 1 and 2 (OOL03 Open Land Area outside the RA); a narrow strip of land from Unit 3 to the discharge canal (OOL04- Sump Drain Line Land Area); and the count room building (CRB). The area enclosed within Unit 3's RA fence is survey area NOL0, which consists of roughly 7,617 square meters (m2). Survey area NOL0 is comprised of soils, engineered materials, gravel, and sand, and contains all the structures, systems, and components supporting Unit 3. Survey area NOL0 was affected by events such as liquid radioactive waste (LRW) concentrator steam condensate leakage, overflow of the condensate tank, overflow of LRW concentrator to the concentrated waste tank (CWT) vault, overflow of condensate demineralizers, contamination at railroad gate and the railroad French drain, drum spill on the radioactive waste treatment building, leakage from the SFP, and wet and dry deposition from stack releases. The licensee identified various transport pathways from survey area NOL0 via infiltration from the spills, drains, and the SFP to subsurface soils and via wet and dry deposition of radioactive effluents. The licensee conducted an expansive characterization of soils and sediments in 1997 and reported the results for Co-60 and Cs-137 concentrations that ranged between non detect (ND) amounts to 39.2 pCi/g of Cs-137 at a depth of 1.5 m (5 feet (ft)) and 3.6 pCi/g of Co-60 at a depth of 0.15 m (0.5 ft). The licensee reported that random sampling for HTD isotopes consisted of analysis of Am-241 in 59 samples, and Cm 243/244/245/246, Pu 238/239/240/241, and Ni-63 in 18 samples. The licensee observed four of the 59 samples measured Am-241 with the maximum result of 0.26 pCi/g and a minimum detectable activity (MDA) analytical range from 0.06 to 0.33 pCi/g. The licensee observed only one of the 18 samples measured Cm-243/244 at 0.08 pCi/g, which was above the MDA of 0.06 pCi/g, and five samples had measurements of Cm-245/246 with a maximum concentration of 0.05 pCi/g, which was above the MDA range of 0.01 to 0.07 pCi/g. The licensee observed only one of the 18 samples measured Pu-238 at 0.14 pCi/g at an MDA of 0.08 pCi/g, and two samples measured concentrations of Pu-239/240 with a maximum of 0.14 pCi/g and an MDA range of 0.01 to 0.10 pCi/g. The licensee did not detect Pu-241 in any samples collected and analyzed from survey area NOL01 and measured Ni-63 in only two of 18 samples with the maximum concentration of 0.26 pCi/g and a MDA analytical range from 0.06 to 0.33 pCi/g. The licensee states that although many samples measured concentrations near detection limits, substantially greater radiological contamination occurred at depths to 3.7 m (12 ft) or greater where events occurred and for this reason expects extensive remediation will occur in survey area NOL01.

The licensee states survey area OOL01, Discharge Canal South, consists of approximately 2,471 m2 of surface area of silt and sand. Survey area OOL01 is the site where circulating water from HBPP Unit 3 discharged prior to entering Humboldt Bay, which affected the canal by routine discharges of radioactive liquids from HBPP Unit 3, as well as overflow of LRW concentrator to the CWT vault. Sediments have silted in to a depth of about 3-4.5 m (10-15 ft) since circulating water flow stopped. The licensee reports that activity in the sediments measured in 1997 prior to silting in vary significantly. Concentrations averaged 8.7 pCi/g for Cs 137 and 1.0 pCi/g for Co 60, and that concentrations at the point of entry of the circulating water into the canal are greater than the remainder of readings. The LTP states that the licensee observed only one detection of Cm-245/246 of all samples analyzed for HTD isotopes, which measured 0.04 pCi/g of Cm 245/246 with an MDA of 0.01 pCi/g. The licensee states that the single Cm detection represents a fraction of a derived concentration guideline levels (DCGL) of 5.62E 04, which would not change the data quality objectives (DQOs) or the classification of the survey area. The licensee performed an additional characterization in 2008 and found that radiological contamination was limited to the top 0.6 m (2 ft) in the sediment.

The LTP describes survey area OOL02-Intake East as open area within the eastern portion of the Intake Canal that contains about 628 m2 of surface area made up mainly of silt and sand that LRW concentrator steam condensate leakage may have affected. The largest sample concentrations measured were 22.4 pCi/g Cs-137 and 5.3 pCi/g Co-60 at depths of 0.5 and 0.15 m (1.7 and 0.5 ft), respectively. The licensee only measured one sample of Pu-238 at 0.13 pCi/g with an MDA of 0.12 pCi/g and one sample measuring Pu-239/240 at 0.22 pCi/g with an MDA of 0.08 pCi/g. Survey area OOL03, Open Land Area, outside of the RA and north of Units 1 and 2, contains about 1,989-m2 of surface area of soils and engineered materials that pavement contamination, remediation activities in 1991, and Unit 2 oil sump and oily water separator contamination affected. The licensee reports only Cs-137 and Co-60 measured primarily in surface soils peaking at 23.7 and 0.5 pCi/g, respectively.

The licensee describes survey area OOL04, the Sump Drain Line Land Area, as a strip of open land area from Unit 3 to the discharge canal that buries the sump drain line beneath it and contains about 458 m2 of surface area consisting mainly of soil. The licensee removed utilities and obstructions in this area while constructing the Humboldt Bay Generating Station (HBGS) in 2008. The HBGS is located on the licensed footprint of the site but is not part of the NRC license. The licensee removed soil from the top of the discharge tubes as it relocated a utility line outside the HBGS footprint area and discovered an access portal (i.e., manhole) to a concrete monolith that connected to the radioactive waste tankage drain line that led toward the discharge canal. The licensee found that the drain line, as well as the concrete monolith, was contaminated with exposure measurements of 20 mrem/hr on the concrete surface. The licensee removed the concrete monolith and most of the drain line toward the Unit 3 RA. Soil samples collected and analyzed in the area measured more than 50 pCi/g. The licensee removed the soil to "near background levels." The LTP describes survey area CRB, the Count Room Building, as having 372 m2 of surface area that may have been contaminated by sample preparation. The licensee built the CRB in 2010 for preparing and analyzing samples and for whole body counting. The licensee has not characterized the survey area CRB, and therefore classifies it as a Class 1 area.

The five Class 2 sites are survey areas OOL05, the Discharge Canal North; OOL06, the Intake Center; OOL07, the NOL Boundary East; OOL08, the NOL Boundary West; and OOL09, the Hazardous Waste Area. The LTP describes survey area OOL05 at the north end of the discharge canal that consists of 556 m2 of surface area primarily of silt and sediment and shows an appreciable reduction of activity concentration. Routine discharges of radioactive liquids

from Unit 3, overflow of LRW concentrator to the CWT vault, and leakage from the LRW concentrator steam condensate leakage could have affected the contamination of this area. The LTP states an earlier characterization conducted in 1998 had elevated Cs-137 concentrations in shallow sediments (i.e., 0.5 m (1.7 ft)) in the native sediment, but significantly less activity in the northernmost end of the canal than the rest of the canal. The licensee collected and analyzed 19 sediment borings to characterize further radiological and chemical concentrations in the discharge canal in 2008. The licensee classified the area as Class 2 b activity because small activity concentrations (i.e., <2 pCi/g) of Co-60 and Cs-137 were measured. The licensee did not measure any of the HTD isotopes above the MDA.

The licensee describes survey area OOL06, the intake center, consisting of 2,047 m2 of surface area of silt and sand that may have been affected by the LRW concentrator steam condensate leakage and deposition of stack releases onto the water surface. The LTP states that Cs-137 less than 1 pCi/g were measured. Survey areas OOL07 and OOL08 border the Class 1 survey area NOL01 on the east and west sides, respectively. Survey area OOL07 contains about 8,326 m2 of surface area consisting of soils and engineered materials. Wet and dry deposition from the stack releases and spills transported from Class 1 survey area NOL01 may have affected survey area OOL07. The LTP states that small activity concentrations (i.e., <1 pCi/g) of Co-60 and Cs-137 were measured and no detections above the MDA of the HTD isotopes were measured at survey area OOL07. Survey Area OOL08 contains 6,837 m2 of surface area of soils and engineered materials that stack release deposition and liquid spills from Class 1 survey area NOL01 may have affected. The LTP states that small activity concentrations (i.e., <1 pCi/g) of Cs-137 were measured at survey area OOL08. Survey areas OOL09, the Hazardous waste area, consists of 1,032 m2 of surface area of soils and engineered materials. The placement of slightly contaminated hazardous waste and spoils in survey areas OOL09 may have resulted in general area contamination. The LTP states that small activity concentrations, <4 and <2 pCi/g of Cs-137 and Co-60, respectively were measured and no detections above the MDA of the HTD isotopes were measured at survey area OOL09. The licensee has classified these areas as Class 2 until further assessed.

The LTP states that the HSA determined that the primary radiological contaminants of concern for the HBPP site are Fe-55, Co-60, Cs-134, Cs-137, Ni-63, Pu-238/241, and Am-241. The licensee states that the more abundant fission and activation products, such as Fe-55 and Co-60, have decayed to 0.1 percent and 1.6 percent, respectively, of their total activity since the plant has been in cold shutdown and SAFSTOR since 1976. The licensee states that Cs-137 and Ni-63 are the most abundant radionuclides in the HBPP inventory and has observed an increase in Am-241 since the shutdown of Unit 3 probably from the beta decay of Pu-241 to Am-241. The LTP states that the radionuclide inventory performed in 1981 did not include analysis for Pu 241, conceivably due to detection limitations. The licensee predicts that the increase of Am-241 should reach 90 percent of its maximum in 2013, approximately 48 years from the date of the last fuel cladding failure, which occurred in1965, and estimates that the maximum will occur in 2038, 73 years after the last fuel failure. The licensee concludes from the HSA that the land and structures near the Unit 3 may require radiological remediation. The LTP states that the Unit 3 nuclear reactor and buildings will require radiological remediation before they are demolished to ensure the licensee does not exceed the offsite dose limits defined in the HBPP's Offsite Dose Calculation Manual (ODCM) and can achieve a final status survey (FSS). The licensee commits to disposing all materials above the DCGLs at an NRC-licensed waste disposal facility. The licensee concludes that migration of surface and subsurface contamination is limited to areas within vicinity of Unit 3 and that HBPP operations will not affect the HBGS facility and the ISFSI, and therefore, will not require radiological remediation.

The LTP describes the radiological site characterization surveys conducted in 1997 and 2008. The licensee conducted the 1997 survey to assess the type and extent of the radiological contamination in sediments and shallow soils at HBPP and to obtain a decision-making basis for developing radiological remediation requirements and cost estimates to plan the decommissioning of Unit 3. The LTP stated that the 1997 survey included sampling of sediments and surface and subsurface soils to a depth of 1.2 m (4 ft) below ground surface (bgs), which the licensee based on a preliminary classification of HBPP soils and sediments prepared by Battelle in 1983. The licensee selected survey instrumentation to ensure that sensitivities were sufficient to detect the expected radionuclides at the minimum detection requirements, and included a sodium iodide (NaI) detector, a gamma scintillator, and plastic scintillator, to conduct qualitative soil measurements, and a high purity germanium (HPGe) gamma spectroscopy detector to identify isotopes.

The LTP states that the licensee conducted the 2008 characterization surveys following MARSSIM guidance to assess the radiological status of the HBPP site, which used data quality objectives (DQO) to establish the necessary requirements and methodologies. HBPP used NaI detectors calibrated with Am-241 and Cs-137 to survey soils and measure area exposures, gas flow proportional counters calibrated with Th-230 and Tc-99 to conduct surface static and scan measurements, and a Zinc Silicon diode (Zn-S) calibrated with Pu-239, Tc-99, I-131, and C-14 to analyze smears collected. The licensee reports that it used this instrumentation and calibration methodology to optimize detection capabilities and accurately characterize the site.

The LTP states that the licensee will continue to characterize the site as decommissioning progresses making additional areas accessible, collecting additional sampling data as needed, and that the licensee will continue to evaluate data as collected to determine the impact on the radioisotopes present, nuclide fractions, and the classification of structures and environmental media. The LTP concludes that 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, decommissioning and decontamination (D&D) activities, estimating radioactive waste types and volumes, and for the development of the DCGLs. The NRC will continue to evaluate, by future in-process and confirmatory inspections, the licensee's activities and how this information will be used in implementing the FSS.

### 3.1.2 Site Characterization - Summary Finding

The NRC staff has reviewed the information in the LTP for the HBPP site according to Section B.2 of the standard review plan (SRP), NUREG-1700 (NRC, 2003). The LTP summarizes the original shutdown and current radiological status of the site. The LTP identifies all locations, where spills, disposals, operational activities, or other accidents and or incidents occurred that could have resulted in contamination within and outside the facility. The LTP describes the areas and equipment that need additional remediation, identifies background activity concentrations and radiation exposure readings used during scoping and characterization surveys, and identifies the survey instruments and supporting QA practices used in the site characterization program. The licensee has sufficiently detailed the status of the HBPP to allow the staff to determine the extent and range of radiological contamination of the HBPP facility structures and site. Therefore, the LTP meets the acceptance criteria as delineated in SRP Section B.2. Based on this review, the staff has determined that the licensee has met the objectives of providing an adequate site characterization as required by 10 CFR 50.82(a)(9)(ii)(A).

## 3.2 Remaining Site Dismantlement Activities

In accordance with 10 CFR 50.82 (a)(9)(ii)(B), and following the guidance of NUREG-1700, "Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans," and Regulatory Guide 1.179, "Standard Format and Content of License Termination Plans for Nuclear Power Reactors," the licensee provided a description of the major remaining dismantlement and decontamination activities as of April12, 2013. The information included those areas and equipment that need further radiological remediation and an estimate of radiological conditions that the licensee may encounter. The LTP included estimates of associated occupational radiation dose and projected volumes of radioactive waste. The licensee provided an overview and describes the major remaining components of radiologically contaminated plant systems and, as appropriate, a description of specific equipment remediation considerations. The LTP also provided information related to the remaining D&D tasks. This information included an estimate of the quantity of radioactive material to be released in accordance with the licensed material disposal requirement of 10 CFR 20.2001, a description of proposed control mechanisms to ensure areas are not re-contaminated, estimates of occupational exposures, and characterization of radiological conditions to be encountered and the types and quantities of radioactive waste. The LTP states that the licensee will decontaminate and dismantle HBPP Unit 3 in accordance with the DECON[4] alternative, as described in NUREG-0586, "Final Generic Environmental Impact Statement" (GEIS) (NRC, 2002). The LTP states that completion of the DECON alternative is contingent upon the licensee's access to one or more low-level waste (LLW) disposal sites. The LTP states that the licensee presently has access to the disposal facilities in Utah, Texas, and Idaho. The licensee is currently conducting D&D activities at the HBPP site in accordance with HBPP procedures, approved work packages providing detailed task instructions, and in coordination with the appropriate Federal and State regulatory agencies. The final state of the HBPP site will be an electrical production facility for approximately 30 more years. The impact of decommissioning activities the licensee will perform will be to reduce residual radioactivity to an amount of 0.25 mSv/yr (25 mrem/yr) and as low as reasonably achievable (ALARA) from all potential pathways to the average member of the critical group (i.e., RESRAD[5]'s residential farmer).

Between August and December 2008, the licensee removed all onsite spent fuel assemblies from the SFP, loaded the spent fuel into five Hi-Star HB casks, and then placed the casks into the HBPP Unit 3 ISFSI; the ISFSI is licensed under a specific 10 CFR Part 72 license. The licensee has a sixth Hi-Star cask available for storing the Greater Than Class C (GTCC) waste that will result when the reactor vessel is dismantled. This sixth cask will also be placed in the ISFSI. The licensee has removed the spent fuel racks, the evaporator and various tanks from the LRW building, the drywell shield plugs, the drywell head¸ and reactor internals from the reactor building, and the main turbine, condenser, and piping from the turbine building, and the turbine building above the subgrade structures.

As of December 31, 2013, the licensee has shipped approximately 6,016 cubic meters (m3) (212,450 cubic feet (ft3)) to various radioactive waste disposal facilities. The licensee estimates the remaining waste is 60,000 m3 (2,118,900 ft3), most of which is very low activity soils, sediments, and concrete debris. This volume of waste exceeds the NUREG-0586 volume of

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[4] The acronym, "DECON," stands for decontamination, one of the three decommissioning options identified in Section 3.2 of NUREG-0586. The DECON option is defined as: "the equipment, structures, and portions of the facility and site that contain radioactive contaminants are removed or decontaminated to a level that permits termination of the license shortly after cessation of operations."

[5] RESRAD: A computer code used to calculate public doses from contaminated soil or surfaces.

18,343 m3 (647,777 ft3) for the reference boiling water reactor. The LTP states that the additional waste is due to the removal of the caisson and the removal of low-level sediments in the discharge canal. The licensee included a supplement to the Environmental Report addressing the environmental impacts from the additional volume of waste generated in Chapter 8 of the LTP.

The LTP Table 3-1 summarizes the major remaining D&D tasks and proposed completion dates that will occur between late 2014 and ending with the FSS in late 2019. Many of these activities are underway with completion dates now delayed from those listed in Table 3-1. The licensee describes the control mechanisms, such as personnel training and barriers, that it will use to control the spread of radiological contamination and personnel exposure. LTP Table 3-2 presents HBPP cumulative site dose estimates for the D&D project. The licensee estimates the total nuclear worker exposure during decommissioning to be less than 86 person-rem, which 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.

The licensee plans to remove all above-grade structures and grade the site except for the Count Room Building, Waste Management Building, Security buildings, administration buildings, Training building, ISFSI, and the HBGS. These buildings will be the only structures to remain onsite at the time of license termination.

The staff has reviewed the LTP against the acceptance criteria in SRP Section B.3. The LTP discusses the remaining tasks associated with D&D, estimates the quantity of radioactive material to be shipped for disposal or processing, describes the proposed control mechanisms to ensure that areas are not recontaminated, and contains occupational exposure estimates and radioactive waste characterization. The licensee has sufficiently detailed data for use in planning further D&D activities and lists the remaining activities that do not require any additional licensing action. Based on this review, the staff has determined that the licensee has identified, in sufficient detail, the remaining dismantlement activities necessary to complete decommissioning of the facility, as required by 10 CFR 50.82(a)(9)(ii)(B) and 10 CFR 50.82(a)(11)(i). Further, the staff has determined that the remaining dismantlement activities can be completed in accordance with 10 CFR 50.59 and will not be inimical to the common defense and security or to the health and safety of the public pursuant to 10 CFR 50.82(a)(10).

### 3.3 Plans for Radiological Site Remediation

In accordance with the requirement of 10 CFR 50.82(a)(9)(ii)(C), the licensee provided its plans for completing radiological remediation of the site. The licensee plans to remediate the site, including structures and systems that remain on the site, to the criteria of 0.25 mSv/yr (25 mrem/yr) for all pathways, which is the unrestricted use criteria specified in 10 CFR Part 20, Subpart E. The licensee evaluated the site after decommissioning using a resident farmer scenario.

The licensee may use remediation techniques, such as washing, wiping, pressure washing, vacuuming, scabbling, chipping, and sponge or abrasive blasting for the structural, metal and concrete surfaces. Scabbling and chipping are mechanical surface removal methods that the licensee intends to use for concrete surfaces. Activated concrete removal 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.

The licensee will remove and dispose of soil contamination above the site specific DCGL as radioactive waste. The licensee may use shovels and back and track hoe excavators as soil remediation equipment. The licensee may use a squared edge excavator bucket design or similar technique when the remediation depth approaches the soil interface region between unacceptable and acceptable contamination, to minimize the mixing of contaminated soils with acceptable lower soil layers as would occur with a toothed excavator bucket. The licensee commits to the use of excavation safety and environmental control procedures to remediate radiologically contaminated soils. The licensee will augment the excavation safety and environmental control procedures with procedural requirements to ensure the licensee maintains adequate erosion, sediment, and air emission controls during soil remediation.

Section 4.3 of the LTP states that the Radiation Protection Program approved for decommissioning is similar to the program that was in place during commercial power operation. The licensee states that contaminated structures, systems, and components were decontaminated in order to perform maintenance or repair actions during power operations and SAFSTOR. The LTP states that the methods the licensee used to reduce personnel exposure to radiation and contamination and to prevent the spread of contamination from established contaminated areas during completed decommissioning activities are the same or similar to the methods the licensee used during reactor operations. The licensee provided its ALARA analysis process in LTP Section 4.4. The licensee’s formulas for calculating the remediation levels conform to the guidance provided in Appendix N of NUREG-1757, Volume 2, “Consolidated NMSS Decommissioning Guidance -Characterization, Survey, and Determination of Radiological Criteria.”

The staff compared the information in the LTP against the acceptance criteria in SRP Section B.4. The LTP discusses in detail how the licensee intends to remediate HBPP to meet the proposed residual radioactivity levels (DCGLs) for license termination. The LTP includes a schedule that demonstrates how and in what time frames the licensee intends to complete the interrelated decommissioning activities. Based on this review, the staff determined that the licensee has met the requirements of 10 CFR 50.82(a)(9)(ii)(C) by providing a detailed discussion of its radiological remediation site plans for the remaining decommissioning activities.

## 3.4 Final Status Survey Plan

### 3.4.1 Final Status Survey Plan Design and Overview

Section 5 of the Humboldt Bay Power Plant (HBPP) License Termination Plan (LTP) discusses the Final Status Survey Plan (FSSP) for the site. Details on the overall scope and purpose of the FSSP are provided in Section 5.1 of the LTP. The FSSP describes the Final Status Survey (FSS) processes to demonstrate compliance with the unrestricted use criteria of 10 CFR 20.1402, and notes in LTP Section 5.1.3 that the FSSP will provide the basis to develop FSS procedures and apply existing procedures to the FSS process. The FSSP also indicates in Section 5.1.3 that an FSS Package will be produced for each survey unit, which details the survey design, survey implementation, and data evaluation for the FSS. A flow diagram for the FSS Process Overview was also provided as Figure 5-1 of the LTP. Survey preparation and survey design are described in Sections 5.1.3.2 and 5.1.3.3 of the LTP, respectively. Section 5.1.3.2 (Survey Preparation) notes that only survey data collected in compliance with approved procedures or as specified in a license amendment request can be used for FSS purposes, and that areas where a FSS has been completed will be isolated or controlled to prevent additional radioactive material from entering the area. Section 5.1.3.2 (Survey Design) indicates that

survey areas will be classified as radiologically impacted or non-impacted, and that impacted survey areas will be divided into survey units designated as Class 1, 2, or 3 depending on the expected level of residual radioactivity. HBPP also indicates that survey unit size will be based on the dose assessment models in accordance with NUREG-1757, Volume 2 (Consolidated Decommissioning Guidance: Characterization, Survey, and Determination of Radiological Criteria). An overview of survey data collection was provided in LTP Section 5.1.3.4, which noted that technicians will perform measurements per approved procedures and instructions using appropriately calibrated instruments. In some cases, survey units will be selected by the licensee for a Quality Control (QC) survey where replicated measurements will be taken for comparison. Data assessment is described in Section 5.1.3.5 of the LTP, which noted that statistical analyses will be performed in order to compare survey results to investigation levels, and to determine if data are sufficient to demonstrate that survey units meet the unrestricted release criterion. Once analyzed and processed, the licensee will document the FSS results in an FSS Package as described in LTP Section 5.1.3.6, and FSS Reports will be prepared for submittal to the NRC. The licensee also notes in LTP Section 5.1.4 that the FSSP was developed using guidance from 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, Volume 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.

Additional details on individual portions of the HBPP FSS Plan are discussed in the proceeding sections.

### 3.4.2 Final Status Survey Plan Development

Section 5.2 of the HBPP LTP discusses FSSP development and provides a summary of site characterization and dose modeling results applicable to the development of the FSSP. An overview of radiological status of the site is described in LTP Section 5.2.1.1, which noted that a site-specific suite of nuclides potentially present at HBPP had been developed. Additional details on characterization are provided in Section 2 of the LTP, and Section 6 describes the development of the site-specific suite of radionuclides. A dose modeling summary is provided in LTP Section 5.2.1.2, which indicated that guidance found in NUREG/CR-5512, Volumes 1, 2, and 3 was used to develop the dose models. The conceptual model was also developed based upon anticipated site conditions at the time of unrestricted release and is designed to be consistent with guidance on site-specific modeling, as provided in Appendix I of NUREG-1757,

Volume 2. Derived Concentration Guideline Levels (DCGLs) were developed using scenarios described in NUREG/CR-5512, Volume 1. Additional discussion on the licensee's development of DCGLs is provided in 3.5 of this report.

Section 5.2.1.3 of the LTP discusses the development of surrogate ratio DCGLs and notes that they will generally be applied to land areas and materials with volumetric residual radioactivity where constant ratios can be demonstrated to exist. It is also noted in the LTP that Cs-137 is expected to be the surrogate nuclide at the site, and its detection will be used to quantify hard to detect radionuclides based upon the ratio established between Cs-137 and the radionuclides of concern which are difficult to detect. The licensee developed a process to assess the need for surrogate ratios when conducting the FSS via the Data Quality Objectives (DQO) process, details of which are given in Section 5.2.1.3 of the LTP. The general process the licensee plans to use is to determine if hard to detect nuclides are likely to be present based on process knowledge, historical data, or characterization results, and when hard to detect nuclides are likely to be present, to establish a surrogate ratio using a representative number of samples (typically six or more). The equation that will be used to develop surrogate DCGLs is MARSSIM Equation 4-1 and is provided in Section 5.2.1.3 of the LTP. In order to verify that surrogate ratios remain valid for the FSS, the licensee commits to either performing hard to detect analyses on post-remedial samples or analyzing at least 10% of the FSS samples for hard to detect radionuclides. The licensee has also committed to store all FSS samples on-site until survey units are approved by the NRC in case additional analyses are necessary. In order to address situations where multiple radionuclides are present at concentrations exceeding 10% of their respective DCGLs, Section 5.2.1.4 of the LTP provides the methodology for gross DCGL development. The associated gross DCGL equation (as based on MARSSIM Equation 4-4) is also provided.

Section 5.2.2 of the LTP describes the classification of areas. Section 5.2.2 noted that structures and open land areas were classified using site characterization results and in accordance with guidance in Appendix A of NUREG-1757, Volume 2, and Section 4.4 of NUREG-1575. As such, the site has been divided into non-impacted and impacted areas. Impacted areas are further divided into Class 1, 2, and 3 areas (as defined in NUREG-1757, Volume 2) depending on the anticipated level of contamination. The licensee also notes that if the available information is not sufficient to designate an area as a particular class, that area will be further characterized or given the most restrictive (Class 1) designation. Areas considered to be borderline between two classes will also be given the more restrictive of the two. In the event that reclassification of an area is necessary during the decommissioning and FSS processes, the licensee indicated in LTP Section 5.2.2 that reclassification to a more restrictive area will be performed in accordance with LTP Section 5.3.6.4. A reclassification to a less restrictive designation will require a 14 day notice prior to commencement of the survey, as noted in Section 1.6 of the LTP. Initial classifications of structural surfaces and land areas are provided in LTP Section 5.2.2.3 and Table 5-2. It is also noted that, for operational efficiency, survey areas listed in LTP Table 5-2 may be subdivided into multiple survey units with the same original classification, unless reclassification is done in accordance with the LTP.

Section 5.2.3 of the LTP describes the processes to establish survey units within the larger survey areas. The licensee intends to select survey unit sizes based on area classification, survey execution logistics, and applicable regulatory documents. Table 5-3 of the LTP lists typical survey unit sizes for structural surfaces and open land areas, and these are consistent with the guidance in NUREG-1575. In the event that survey unit sizes larger than those listed in LTP Table 5-3 are used, the licensee commits to providing a technical evaluation and justification in the FSS Package.

Access control measures are described in LTP Section 5.2.4, which indicated that some surveys may be completed in parallel with dismantlement activities. As such, access controls and measures to prevent recontamination of an area are planned. Examples of such access controls provided in LTP Section 5.2.4 include personnel training, postings, and periodic surveillance surveys. Additionally, the licensee's Site Closure Manager will ensure that all decommissioning activities in areas adjacent to an area to be isolated for FSS, or that could otherwise impact an area ready for FSS, are complete or have no potential to spread residual radioactivity to the isolated area. Sections 5.2.4.3 and 5.2.4.4 of the LTP describe walkdown and transfer of control measures. Once a walkdown has been completed to ensure that an area has been left in a state ready for FSS, and no survey impediments exist, control of the area is transferred from HBPP Radiation Protection (RP) to the FSS group. Additional information on specific measures for isolation and access controls are provided in LTP Section 5.2.4.4. Examples of the planned measures include personnel training, installation of access control and environmental control barriers, locking of entrances to surveyed areas, installation of tamper evident devices at entry points, periodic surveillance/inspections, and postings of areas.

### 3.4.3 Survey Design and Data Quality Objectives

Survey design and Data Quality Objectives (DQOs) for decommissioning at HBPP are described in Section 5.3 of the LTP. The methods and data required to determine the number and location of measurements or samples in each survey unit and the coverage percentage for scan surveys are described. The LTP states in Section 5.3 that the licensee will consider Type I and II errors, scan survey coverage, sample size determination, instrumentation and required Minimum Detectable Concentrations (MDCs), sample location, and DCGL/$DCGL_{EMC}$ determination as FSS surveys are designed. Section 5.3.1 of the LTP specifically outlines the DQO process, which is based on guidance in MARSSIM, Appendix D. The steps that will be utilized in that process are as follows: state the problem, identify the decision, identify the inputs to the decision, define the study boundaries, develop a decision rule, specify limits on decision errors, and optimize the design for obtaining data.

Radiological scan coverage for FSS survey units is discussed in LTP Section 5.3.2, and it is noted that the coverage is consistent with guidance in NUREG-1757. As such, Class 1 areas will have 100% scan coverage, Class 2 areas will have 10-100% coverage, and Class 3 areas will have judgmental (1-10%) coverage. Sample size determination is described in LTP Section 5.3.3 and follows processes described in NUREG-1757, Volume 2, Appendix A. The licensee notes in Section 5.3.3.1 that appropriate tests such as the Sign Test and the Wilcoxon Rank Sum (WRS) Test will be used depending upon the level at which the contaminant of interest is also found in background. As noted in Section 1.6 of the LTP, a change in the statistical test applied to other than the Sign Test or Wilcoxon Rank Sum Test requires prior NRC approval. This change control process is specified in the license condition added as part of this license amendment and is consistent with guidance in NUREG-1700, "Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans." The methodology to establish decision errors uses a MARSSIM Scenario A Null Hypothesis ($H_0$) that the survey unit does not meet the release criteria and an Alternate Hypothesis ($H_A$) that the survey unit does meet the release criteria, as described in Section 5.3.3.2 of the LTP. Type I and II decision errors are also described in LTP Section 5.3.3.2, which noted that the probability of making a Type I error ($\alpha$) will be set to 0.05 per guidance in NUREG-1757, Volume 2. The licensee plans to set the probability of making a Type II error to 0.05 initially, but may modify this value as necessary after considering 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 limit.

The calculation of other values necessary for sample size determination, such as the relative shift, the lower bound of the gray region, and standard deviation are all described in Section 5.3.3 of the LTP. The required number of samples for each survey unit will be determined using MARSSIM Table 5-3 and Table 5-5 for the WRS Test and Sign Test, respectively. Section 5.3.3.3.5 of the LTP indicates that sample size adjustments will be necessary if a Scan MDC exceeds the $DCGL_w$. In that case, sample size will be adjusted using NUREG-1757, Equation A-8 (listed as Equation 5-3 of the LTP). If the WRS Test is to be used, a background reference area is required. Section 5.3.4 of the LTP discusses the development of background reference areas and notes that reference area measurements will be collected using methods and procedures required for Class 3 final status survey units. It is also noted that for soil, reference areas will have a soil type as similar to that of the survey unit as possible. For structure surveys where survey units contain markedly different backgrounds, a reference area with similar materials will be selected. In the event that one structural material is predominant, or there is no large variation in background among materials, a background from an area with a single structural material may be chosen when it is demonstrated that underestimation of residual radioactivity will not occur. If significant variations are encountered among reference areas, the licensee intends to evaluate the areas and define the background concentration using guidance from NUREG-1757, Appendix A, Section A.3.4 (Kruskal-Wallis test) and other statistical guidance.

Section 5.3.5 of the LTP describes the methods that the licensee will use to develop a grid for systematic sampling and to determine specific sampling locations. For Class 1 and 2 areas starting points for systematic sampling will be randomly chosen. For Class 3 areas, all sample locations will be randomly selected. Grid spacing for Class 1 and 2 areas will utilize a triangular grid pattern per MARSSIM Equation 5-5 (Equation 5-4 in the LTP), and sample locations for Class 3 areas will be defined using a random number generator or Visual Sample Plan (VSP) software.

The development of investigation levels, or radioactivity levels which warrant further investigation, along with the elevated areas test are described in LTP Section 5.3.6. Those investigation levels are provided in LTP Table 5-5 and are shown the table below:

**HBPP FSS 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 $DCGL_W$ |

Section 5.3.6.1 of the LTP notes that locations where residual radioactivity potentially exceeds the investigation levels will be documented and marked for further investigation. It is further noted that results of the investigation may also result in a reclassification of all or part of a Class 2 or 3 survey unit, or survey units may be combined with adjacent areas of similar classification. In either case a resurvey would be performed.

Section 5.3.6.3 of the LTP addresses elevated measurement comparison, which is applied to Class 1 surveys where one or more scan or measurement exceeds the investigation level. These locations will be subject to additional surveys in order to determine the size and extent of the elevated area. Results will be compared to a DCGL for elevated measurements ($DCGL_{EMC}$), which is an *a priori* calculation based upon the $DCGL_W$ and an area factor (which is defined as the multiple of the $DCGL_w$ that is permitted in a given area of elevated radioactivity without remediation). Equation 5-5 of the LTP provides this relationship. Tables 5-6 and 5-7 provide area factors calculated for soil areas and building surfaces, respectively. Actual results within affected survey units will be used to determine *a posteriori* area factors and $DCGL_{EMC}$ values, and these will be used for compliance purposes.

Section 5.3.6.3.2 of the LTP noted that DCGLs for HBPP embedded and buried piping will be in accordance with HBPP Technical Basis Documents (TBD) and that the HBPP embedded piping DCGL TBD will be submitted to the NRC for approval prior to implementation. Additional information on DCGL development for embedded pipes was provided during the RAI process, and this is further discussed in 3.5.5 of this SER.

Remediation and reclassification is discussed in Section 5.3.6.4 of the LTP, where it is noted that Class 1 areas including residual radioactivity above the $DCGL_{EMC}$ will be remediated. In the event that an individual measurement in a Class 2 survey unit exceeds the $DCGL_W$ HBPP intends to evaluate whether all or part of a survey unit should be reclassified to Class 1 and resurveyed. Individual measurements in Class 3 areas which exceed 50% of the $DCGL_W$ will trigger an evaluation to determine whether all or part of the survey unit should be reclassified to a Class 2 area. Table 5-8 of the LTP provides the framework for the investigative actions necessary to evaluate areas for reclassification and resurvey. Section 5.3.6.5 of the LTP provides details on resurveys after reclassification and notes that if a survey unit is reclassified to a more restrictive classification or if remediation activities were performed, a resurvey will be performed in accordance with approved procedures.

### 3.4.4 Survey Methods and Instrumentation

Survey methods and instrumentation are discussed in Section 5.4 of the LTP, which noted that FSS measurements will include surface scans, direct surface measurements, and gamma spectroscopy of volumetric materials. The licensee also proposes in LTP Section 5.4.1 that “Advanced Survey Technologies” not specifically described in the LTP may be used, and that 30 days’ notice will be provided to the NRC to provide an opportunity to review the associated basis document. When offsite laboratory analyses are required, they will be conducted in accordance with Section 5.8 of the LTP and the FSS Quality Assurance Project Plan (QAPP). Analytical methods, for both offsite and onsite analyses, will be administratively established to detect levels of radioactivity between 10-50% of the DCGL values.

FSS measurements planned for structures at the HBPP site are discussed in Section 5.4.2 of the LTP and will include scan surveys, direct measurements, and volumetric sampling as necessary. Scan surveys are described in LTP Section 5.4.2.1 and will be performed to locate small elevated areas of residual radioactivity. It is also noted that measurements will typically

be performed at a distance of 1 cm or less from the surface at a nominal speed of 5 cm/s using hand-held instruments. Direct measurements are described in LTP Section 5.4.2.2. These will typically be one minute counts with the detector placed near the surface, though it is noted that count times may need to be adjusted to ensure that an adequate detection level is achieved.

Activated concrete and volumetric concrete measurements are discussed in LTP Sections 5.4.2.3 and 5.4.2.4, respectively. Activated concrete that does not meet the FSS criteria will be removed and shipped to an appropriate burial site. The licensee also notes in Section 5.4.2.4 of the LTP that volumetric sampling of concrete may be necessary when the detection capabilities of gross beta measurements are limited. In this case, a surface layer will be removed and analyzed, and the results will be converted to the appropriate surface activity units. Direct measurements of underlying surfaces would then be performed after the coating is removed. It is also noted that the thickness of any layer to be removed as a sample should be consistent with the HBPP site model and the DCGLs. The licensee assumes that a depth of less than 10 mm will be used for this purpose and that for the radionuclides of concern, this depth would provide a minimum degree of shielding.

FSS sampling for soils is described in Section 5.4.2.5 of the LTP. Soil scanning is specifically discussed in LTP Section 5.4.2.5.1, while volumetric sampling of soils is discussed in LTP Section 5.4.2.5.2. For scans, open land areas will be scanned for gamma emitting radionuclides, typically using sodium iodide detectors, at a distance of 2.5 to 7.5 cm from the surface and at a speed of 0.5 m/s. Each square meter will be traversed three times. Volumetric samples of soils will be analyzed by gamma spectroscopy. Surface soil samples will typically be taken at a depth of 0 to 15 cm, while areas of potential subsurface contamination may require sampling from 15 cm to 1 m. It is also noted that sodium-iodide or an in situ object counting system (ISOCS) may be used to identify the potential presence of subsurface contamination (below 15 cm in depth) which would trigger an investigation.

Section 5.4.2.5.2 of the LTP states that if contamination at a significant fraction of the DCGL is identified, the licensee will perform confirmatory analyses and an investigation of the area. The licensee stated that subsurface soil sampling will be performed in accordance with Section G.2.1 of NUREG-1757, Volume 2. The licensee also acknowledges that "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," and that "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." NRC staff recognizes that guidance on subsurface sampling is limited, and that Section G.2.1 of NUREG-1757, Volume 2 does not provide an exhaustive methodology to address subsurface sampling. There are potential complications associated with subsurface sampling for FSS compliance purposes, such as limited scanning ability and the evaluation of elevated areas at greater depths. As such, Section G.2.1 of NUREG-1757, Volume 2 notes that "generic guidance has not yet been developed for performing an EMC for subsurface samples; therefore, licensees should discuss this matter with NRC staff on a case-by-case basis." NRC staff agrees with the licensee's commitment to consider the potential complications of using subsurface sampling/assessment for compliance purposes during the DQO process and to provide additional details to the NRC to ensure that sampling and evaluation methods are appropriate.

An alternative survey plan in excavations is discussed in Section 5.4.2.5.3 of the LTP, as there may be instances where deep excavations are made and it is necessary to remove

radiologically contaminated soils along with both clean and contaminated foundations and underground utilities. In these cases, shoring or trench boxes will be required to allow safe personnel access. Since 100% surface scans are not possible in Class 1 areas in this situation, the licensee proposes the following assessment methodology:

> 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 Technical Basis Document will be developed for the bulk assay system and submitted to the NRC prior to being used.
>
> FSS will be performed on the bottom of the excavation prior to any backfill.
>
> Side 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.

NRC staff recognizes that there may be safety concerns associated with surveys in excavated areas requiring trench boxes or shoring and that typical FSS practices may not be possible. As survey methodologies in these areas may be adjusted on a case by case basis, the licensee should be able to demonstrate that the chosen sampling methods are representative of soils behind trench walls/shoring and that the extent of areal contamination is understood. This could be subject to in-process NRC review, inspections, confirmatory surveys, etc.

Section 5.4.3 of the LTP discusses specific survey considerations for areas that may require a more unique sampling approach. These areas are addressed as follows:

Section 5.4.3.1 states that for pavement-covered areas and shallow concrete slabs, surveys will be based on soil survey unit sizes, and applicable DCGLs will be the soil DCGLs. Concrete/asphalt samples will be taken along with soil samples underneath.

Bulk materials are discussed in LTP Section 5.4.3.2, which noted that some excavated soils may be reused on site after an appropriate survey. The licensee proposes to either survey excavated soils in 6 inch lifts using methodologies consistent with the MARSSIM FSS process or to use bulk survey methods, such as an in-situ object counting system (ISOCS), provided the methodology can accurately assess the materials. In the first case, the licensee indicated that excavated soil will be characterized to determine suitability for transport to an area dedicated for excavated soils and that soils not containing residual radioactivity above the DCGL will be relocated to a dedicated soil evaluation area. Soils will then be leveled to a depth of six inches, and a Class 1 FSS will be conducted with soil measurements averaged over the total depth of soil. The sample/measurement density will be equal to that required for a surface soil survey of the same volume, and surface scanning/volumetric analyses will be directly compared to the DCGL values. Any location determined to contain residual radioactivity above the DCGL will be investigated and remediated as necessary.

The licensee will institute controls to prevent mixing of soils from different survey units prior to evaluation. Soils satisfying the criteria for unrestricted release will be stockpiled for use as onsite backfill and will be available for usage in site areas with the same (or more restrictive) survey unit classification. For example, material from Class 1 areas can only be reused in a Class 1 area, while Class 2 materials could be reused in a Class 2 or 1 area, and Class 3 material could be used in any onsite excavation backfill. Additionally, the licensee indicated in Section 5.4.3.2 of the LTP that stockpiles will be maintained under the Stormwater Pollution Prevention Plan Best Management Practices, which requires the piles to have a lower and upper cover and be waddled in the middle. With regard to the second proposed survey method, bulk survey, the licensee indicated that a GARDIAN-III measurement system may be utilized. This unit includes a total of six High Purity Germanium (HPGe) detectors that have been characterized for ISOCS measurements to allow modeling of large volume geometries for accurate quantification of results. The detectors are housed in two semi-trailers parked in a parallel configuration and two large volume plastic scintillation detectors are mounted outside on the in-board ends of each trailer. NRC staff notes that either method must be able to adequately assess soils to the same rigor of a MARSSIM FSS. As such, materials surveyed via lifts or bulk survey methods will be subject to the same NRC review, inspection, and confirmatory surveys associated with all site FSS survey areas. NRC staff encourages the licensee to discuss proposed bulk material survey methodologies with the NRC and to consider the capabilities of the methods during the FSS planning process.

Embedded and buried piping are discussed in Section 5.4.3.3 of the LTP, which noted that separate FSS survey plans will be developed for embedded/buried piping, including survey unit DQOs. The LTP also states that accessible internal surfaces of pipes will be surveyed in the same manner as structural surfaces. Scale and sediment samples will also be obtained, as well as smear or swipe samples. Internal surface activities will be compared to building surface DCGLs. The licensee also commits to develop specialized piping DCGLs and a technical basis document if samples indicate that a survey unit will be failed. The licensee plans to grout all remaining embedded and buried piping after survey, unless it is part of an active system (i.e., drainage piping).

Cracks, crevices, wall-to-floor interfaces, and small holes are discussed in LTP Section 5.4.3.4, including surveys for surface contamination on irregular structure surfaces. Where no remediation has occurred and residual radioactivity above background has not been detected, the licensee indicates that surface blemishes will be assumed to have the same level of residual activity as adjacent surfaces. In instances where remediation has occurred or residual radioactivity above background has been detected, the licensee intends to take a representative sample of contamination from within the crack or crevice or adjust instrument efficiencies accordingly. If the depth of contamination or other factors do not allow for a justifiable correction to instrument efficiencies, volumetric samples will be collected and results can be converted to a surface activity equivalent value. The licensee plans to survey accessible portions of irregular structure surfaces in the same manner as other structure surfaces, and plans to judgmentally scan these areas in survey units which require less than 100% coverage.

Paint covered surfaces are discussed in Section 5.4.3.5 of the LTP, which stated that painted surfaces will be evaluated prior to the start of an FSS for that survey unit. If activity is suspected beneath painted surfaces, coatings will be removed prior to a survey. The licensee also noted that source efficiency corrections, per NUREG-1507, may be applied to measurements when a coating thickness can be determined with certainty.

Section 5.4.3.6 of the LTP addresses exterior surfaces of building foundations. The licensee intends to use the HSA and other pertinent records to determine the potential for contamination on the exterior of below-grade/sub-surface foundations. The licensee proposes to evaluate these exterior surveys via core borings through foundations or walls, taking soil samples at locations with a high potential for migration of contamination to sub-surface soils, or gamma well logging in soils next to building exteriors.

Groundwater is discussed in Section 5.4.3.7 of the LTP, which noted that assessments of residual activity in groundwater will be performed via monitoring wells for both deep and shallow depths. Data collected will be used to ensure that residual radioactivity in groundwater remains below the U.S. Environmental Protection Agency (EPA) maximum contaminant levels (MCLs), and thus represents a small fraction of the dose limits for unrestricted release of the site.

Section 5.4.4 of the LTP discusses instrumentation that will be used for FSS purposes. Section 5.4.4.1 further discusses instrument selection for both direct measurements and scan measurements. For direct measurements, the licensee plans to use instrumentation with MDCs which are less than 50 percent of the DCGL and scanning instruments for Class 1 areas will at least be able to detect radioactive material at the $DCGL_{EMC}$. Typical FSS instrumentation is provided in Table 5-9 of the LTP, and typical detection sensitivities are provided in Table 5-10. Instrumentation calibration and maintenance is discussed in LTP Section 5.4.4.2, which noted that instruments and detectors are calibrated by the licensee for the radiation types and energies of interest at the site, and that approved suppliers will calibrate instruments as necessary per their approved Quality Assurance Program. Alternatively, calibrations may be performed in accordance with approved licensee procedures at HBPP or Diablo Canyon Power Plant (DCPP). Radioactive sources used for calibration will be traceable to the National Institute of Standards and Technology (NIST). Instrument response checks are described in LTP Section 5.4.4.3, which noted that response checks will be performed daily, both before and after an instrument is used. An acceptable response for field instrumentation is plus or minus 20 percent of the established check source value. Instruments failing response checks will be marked as “Out of Service,” and measurements between the last acceptable check and the failed check will be evaluated to determine whether or not they should remain in the data set.

Section 5.4.4.4 of the LTP discusses MDC calculations related to the instruments and techniques for FSS. The MDC for static or direct measurements is described by Equation 5.8, and the Scan MDC for structural surface beta-gamma scans is described by Equation 5-9. Both equations are based on guidance from NUREG-1507 (Minimum Detectable Concentrations with Typical Radiation Survey Instruments for Various Contaminants and Field Conditions). Section 5.4.4.4.3 of the LTP describes methods that will be used to determine total efficiency as it relates to Scan MDC determination (total efficiency is described by the product of instrument efficiency and source efficiency). The LTP notes that the practical application of choosing the proper instrument efficiency may be determined by averaging the surface variation (surface peaks and valleys narrower than the length of the detector) and adding 0.5 cm (the spacing that should be maintained between the detector and the highest peaks on a surface). Table 5-11 of the LTP describes source to detector distance effects on instrument efficiencies for alpha and beta emitters. Section 5.4.4.4.3 further notes that source efficiencies may either be derived empirically or from guidance in ISO 7503-1 and describes weighting of efficiencies which may be used when a known ratio of multiple radionuclides exists. Structural surface Scan MDCs for alpha radiation are discussed in LTP Section 5.4.4.4.4, which noted that a detection probability calculation using Poisson summation statistics is more useful for alphas. As such, the licensee presents potential detection probabilities based upon guidance in MARSSIM Section 6.7.2.2 and Appendix J and as noted in Table 6.8 of MARSSIM (Probability of Detecting 300 dpm/100

cm$^2$ of Alpha Activity While Scanning with Alpha Detectors Using an Audible Output). The licensee notes that scan speeds may be adjusted if lower MDCs are required. Open land area gamma scan MDCs are described in Section 5.4.4.4.5, which noted that Stage 1 and 2 surface scanning processes for land areas will be considered as defined in NUREG-1507 and that an *a priori* determination of scanning sensitivity will be performed to ensure that the measurement system is able to detect concentrations of radioactivity at levels below the release limit. The licensee provided equations for Scan MDCs of open land surface scanning with a desired performance level of 95% true positive and 60% false positive. Equations 5-11 and 5-12 of the LTP describe the corresponding Minimum Detectable Count Rate (MDCR) calculations that the licensee will use in the determination of land area gamma Scan MDCs. The MDCR values can be used, along with manufacture count rate to exposure rate ratios to determine the minimum detectable exposure rate. The minimum detectable exposure rate will then be used to determine the Scan MDC by modeling a specified small area of elevated activity using MicroShield and converting the exposure rate to an MDC in pCi/g.

High Purity Germanium (HPGe) analysis is described in Section 5.4.4.4.6 of the LTP. The HBPP onsite chemistry laboratory maintains gamma isotopic spectrometers calibrated to various sample geometries. HPGe instruments are calibrated using NIST traceable standards, and count times are set to meet a maximum MDC of 10% of the appropriate HBPP DCGL.

Pipe survey instrumentation is described in LTP Section 5.4.4.4.7, and the licensee noted that accessible portions of any remaining embedded piping will be surveyed to ensure that residual radioactivity is less than the applicable DCGL. The licensee proposes to use scintillation detectors and/or Geiger-Mueller arrays for this purpose. The licensee commits to survey piping, which has been designated as Class 1 for FSS purposes, with 100 percent coverage. This will be accomplished by surveying piping in 1 foot intervals and inaccessible portions will be made accessible by cutting access ports in the piping. An evaluation will be performed to determine necessary survey coverage for Class 2 and 3 areas.

### 3.4.5 Data Collection and Processing

Section 5.5 of the LTP describes data collection, review, validation, and record keeping requirements for the FSS. Sample handling and record keeping is described in Section 5.5.1 of the LTP, which noted that chain-of-custody records will accompany each sample from collection to obtaining final results. Each survey unit will also have an associated document package that covers the design and field implementation of the survey requirements. Data management is described in LTP Section 5.5.2. Survey data will be collected at several points during the data life cycle and will be evaluated throughout the survey process. QC replicate measurements will not be used as FSS data. The licensee also noted that measurements performed during turnover and investigation surveys can be used as FSS data if they are performed to the same requirements as FSS data and the following conditions are met: 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; and the data collection and design are 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).

Data verification and validation are described in LTP Section 5.5.3. The licensee will review FSS data prior to data assessment to ensure they are complete, fully documented, and technically acceptable. At a minimum, the following items will be considered to determine data acceptability:

- 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.
- 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 in accordance with LTP 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.
- 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.

Several different methods of graphical data review are presented in Section 5.5.4 of the LTP, which can be used to identify patterns, relationships, or possible anomalies. Posting plots are described in LTP Section 5.5.4.1 and will be used to identify spatial variability in data. Frequency plots are described in LTP Section 5.5.4.2 and will be used to examine the general shape of data distributions. Contour and 3-D Surface Plots are discussed in Section 5.5.4.3 of the LTP, and these may be used to represent graphically a trend in collected survey data.

### 3.4.6 Data Assessment and Compliance

Data assessment and compliance is described in Section 5.6 of the LTP. Section 5.6.1 describes the criteria to perform FSS data assessments to support a determination to release the survey unit, and evaluation matrices to quickly assess whether or not a survey unit meets the release criteria are described in LTP Tables 5-14 and 5-15 as follows:

**HBPP 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 |

**HBPP 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 |

Section 5.6.1.1 of the LTP further notes that, when required by the measurement results, one of four non-parametric statistical tests will be performed on survey data: WRS Test, Sign Test, WRS Test Unity Rule, or Sign Test Unity Rule. Survey data will also be evaluated against the EMC criteria, in accordance with NUREG-1757, Volume 2, and as described in Section 5.3.6.3 of the LTP.

Details on the statistical tests for compliance are provided in Sections 5.6.1.2 and 5.6.1.3 of the LTP. The WRS test or the WRS Unity Rule (per NUREG-1505, Chapter 11) may be used when the radionuclide of concern is in background or measurements are not radionuclide specific. The Sign Test and Sign Test Unity Rule will be used when the radionuclide of concern is not present in background, or it is present at an acceptably low fraction relative to the $DCGL_w$. If a radionuclide is present in background, it would be included in the results as if it were due to plant operations.

Multiple radionuclide evaluations are described in Section 5.6.2.1 of the LTP. It is noted that the unity rule will be applied, in accordance with NUREG-1505, Chapter 11. LTP Equation 5-13 demonstrates the application of the unity rule using gamma results and the applicable DCGL values. If a surrogate DCGL is being used, it will be appropriately adjusted before inclusion in the unity rule calculation. LTP Section 5.6.2.1 also notes that if the application of the WRS or Sign Test is necessary these tests will be applied using the unity rule equivalent results with a DCGL of 1.0 in a similar fashion to an example provided in Section 11.4 of NUREG-1505.

Section 5.6.2.2 of the LTP discusses elevated measurement comparison evaluations and notes that during FSS, areas of elevated activity may be detected and they will be evaluated individually and in total for compliance purposes. Elevated areas will be compared to the appropriate $DCGL_{EMC}$ value calculated for the size of that elevated area. If individual elevated areas pass, then all elevated areas will be combined and evaluated under the unity rule. LTP Equation 5-14 describes the unity rule analysis for elevated areas and notes that a separate term of the equation will be added for each elevated area identified in a survey unit.

Data conclusions are discussed in LTP Section 5.6.3. Two conclusions will be made based upon the results of the statistical tests, including the application of the EMC. The first possible conclusion is that the data provide statistically significant evidence that residual radioactivity in a survey unit does not exceed the release criterion. The second possible conclusion is that the survey unit fails to meet the release criterion and that data do not conclusively show that residual activity is less than the release criterion. In this case, the licensee will analyze the data to determine the reason for failure. Some potential reasons provided in the LTP are:

- 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).

Section 5.6.3 of the LTP indicates that retrospective power analyses will be performed for each HBPP survey unit using methods from MARSSIM Appendix I.9. LTP Section 5.6.3 also notes that if the power is insufficient due to too few measurements, then additional samples may be collected as directed by procedure. NRC staff notes that two-stage or double sampling is not usually expected (nor is it encouraged) when the DQO process is used, as in the MARSSIM.

This is because the Type II error and the power desired are explicitly considered in the survey design process. It is noted in NUREG-1757, Volume 2, Section A.7.5 that double sampling refers to the case when the survey unit design is a one-stage design, but allowance is made for a second set of samples to be taken if the retrospective power of the test using the first set of samples does not meet the design objectives. Use of either method should be considered as part of the DQO process when developing the design of the FSS. As such, the licensee should refer to guidance in NUREG-1757, Volume 2, regarding double (or two-stage) sampling and should consider such cases in the DQO process for FSS planning.

Section 5.6.3 of the LTP further notes that if failure of the FSS statistical tests was due to the presence of residual radioactivity above the release criterion, the survey unit must be remediated and reclassified as necessary. An investigation and re-initiation of the FSS process will be initiated if failure is due to inadequate design or implementation. Section 5.6.4 notes that the goal of the FSS is to ultimately demonstrate compliance with the unrestricted release criteria as stated in 10 CFR 20.1402. In the event that survey measurements do not meet these criteria, an investigation will be performed to evaluate survey designs, instrumentation use, and calculations, as necessary.

### 3.4.7 Final Status Survey Report

The licensee discussed the Final Status Survey Report (FSSR) in Section 5.7 of the LTP. The FSSR will include the survey results and a brief operating history. Operating history is discussed in LTP Section 5.7.1, and this history will be provided to substantiate the basis for survey unit classifications and the level of intensity for the FSS. The FSSR content is further discussed in Section 5.7.2 of the LTP. The licensee indicated that survey results will be described in a written report for each survey area, which will include information such as the number and type of measurements, basic statistical quantities, and statistical analysis results. LTP Section 5.7.5 also provides the following topics that will, at a minimum, make up the format of the FSSR:

- 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.

The FSSR will also include the survey unit specific or generic ALARA evaluation as well as any investigation performed, regardless of whether or not the survey unit failed.

3.4.8 Final Status Survey Program Quality (QAPP)

Quality assurance plans and programs for the FSS are discussed in LTP Section 5.8. Section 5.8.1 provides a brief description of applicable HBPP quality programs associated with the FSS Quality Assurance Project Plan (QAPP). The HBPP 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 elements of the FSS QAPP are further noted in LTP Section 5.8.1 as follows:

- Project Management and Organization (LTP Figure 5-2 provides an organizational chart of the projected HBPP License Termination Organization.)
- Program Controls
- Design Controls
- Procurement Document Control
- Instructions, Procedures, and Drawings
- Document Control
- Control of Purchased Material, Items, and Services
- Control of Special Processes
- Inspections
- Control of Measuring and Test Equipment
- Handling, Storage and Shipping
- Control of Nonconformance
- Corrective Action Program
- Records
- Audits

### 3.4.9 NRC Staff Summary Evaluation of HBPP Final Status Survey Plan

NRC staff has reviewed Chapter 5 of the HBPP LTP which provides a FSSP based upon guidance in NUREG-1575, NUREG-1505, NUREG-1507, NUREG-1700, NUREG-1757, and Regulatory Guide 1.179. The FSSP meets the requirement for detailed plans for the final radiation survey in accordance with 10 CFR 50.82(a)(9)(ii)(D), and will allow the licensee to comply with the survey and monitoring requirements of 10 CFR 20.1501(a) and (b). In accordance with NUREG-1700, Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans, the LTP adequately describes the final radiation survey plan for demonstrating that the HBPP Unit 3 plant and site will meet the proposed release limits for unrestricted release, as described in 10 CFR 20.1402.

## 3.5 Compliance with Radiological Criteria for License Termination

### 3.5.1 Radiological Criteria for License Termination

In its May 2013 LTP submission, as supplemented, the licensee has requested that the LTP for HBPP Unit 3 be approved under the unrestricted use criteria of 10 CFR 20.1402. Section 20.1402 requires that residual radioactivity levels that are distinguishable from background remaining at the site at the time of license termination cannot result in a total effective dose equivalent (TEDE) to an average member of the critical group that exceeds 25 mrem/yr, including that from groundwater sources of drinking water, and that the residual radioactivity must be reduced to levels that are as low as reasonably achievable (ALARA). As described in its LTP, the licensee performed dose assessments to establish radionuclide specific derived concentration guideline levels (DCGLs) for contaminated soil, building surfaces, and piping corresponding to the 25 mrem/yr criteria.

### 3.5.2 Identification of Potential Radionuclides of Concern

The analysis performed to determine the potential radionuclides of concern at the HBPP Unit 3 is described in the document “Radiological Selection for DCGL Development, Revision 1”. A suite of radionuclides that could be present on the site at the time of shutdown was developed based on NUREG/CR-3474 “Long-Lived Activation Products in Reactor Materials”, NUREG/CR-4289 “Residual Radionuclide Contamination Within and Around Commercial Nuclear Power Plants”, and the HSA.

The inventory of activation products was estimated based on information from NUREG/CR 3474. This inventory was decayed to 09/01/19. Radionuclides that are only produced by activation and that contributed less than 0.1% of the activity were discounted. Additional potential radionuclides were identified based on NUREG/CR-4289. Except for Co-60, radionuclides with half-lives that are less than 5.4 years were not included. Co-60 was included on the list because its estimated inventory at license termination would still be approximately 121 Ci. The HSA also identified Pu-241, Cm-243/244, and Cm-245/246 as present in the waste streams analyzed, so these radionuclides were also added to the list of radionuclides potentially present.

The licensee performed a review of I-129 to determine if this radionuclide should be included in the list of potential radionuclides. They concluded that the activity of this radionuclide contributed less than 0.1% of the total activity. The estimated dose from I-129 was 1.76 x 10-7

mrem for the residential scenario and 1.82 x 10-7 mrem for the occupancy scenario. PG&E also concluded that I-129 concentrations were not found at levels above the MDA in characterization samples at HBPP. For these reasons, PG&E excluded I-129 from the list of potential radionuclides at the HBPP site. PG&E estimated the total dose from the discounted radionuclides for both the residential and building occupancy scenarios and found that the dose from the discounted radionuclides was less than one percent of the calculated total dose.

The site-specific suite of radionuclides identified for HBPP based on the analysis described above includes: Am-241, C-14, Cm-243, Cm-244, Cm-245, Cm246, Co-60, Cs-137, Eu-152, Eu 154, H-3, Nb-94, Ni-59, Ni-63, Np-237, Pu-238, Pu-239, Pu-240, Pu-241, Sr-90, and Tc-99. This suite of radionuclides includes those which would be present in a fuel failure and were identified in the HSA as present or potentially present (i.e., Am-241, Cm 243, Cm-244, Pu-238, Pu-239, Pu-240, and Pu-241).

In Section 6.2.5 of the LTP, PG&E stated that samples will be taken of the soils and building surfaces in areas deemed to have the highest activity present. These samples will be analyzed for all of the radionuclides in the site-specific suite. Radionuclides that are not identified in the analyses may be deselected from the survey, and the potential dose from these radionuclides will be determined based on their MDC values decayed to a license termination date of September 5, 2019.

The NRC staff finds that the approach used by PG&E to identify the potential radionuclides of concern in its LTP is acceptable. The determination of insignificant radionuclides performed by PG&E is consistent with the NRC's guidance in NUREG-1757 Vol 2. This guidance states that radionuclides may be considered to be insignificant as long as the total dose from the radionuclides that were identified as insignificant is less than 10% of the dose criteria. The guidance further states that the dose from the insignificant radionuclides must be accounted for when demonstrating compliance.

### 3.5.3 Development of Soil DCGL Values

PG&E developed soil DCGL values for use in evaluating residual contamination in soil remaining on site following decommissioning.

#### 3.5.3.1 Scenario and Exposure Pathways Evaluated

The soil DCGL values were developed based on the resident farmer scenario. The Humboldt Bay Generating Station, a fossil-fueled power plant, began operating on the site in 2010 and is expected to be operated for at least 30 years, so it is expected that the site will continue being used as an industrial site. However, the resident farmer scenario was conservatively selected by PG&E for the analysis because the dose from this scenario bounds the potential dose from the industrial scenario.

In the resident farmer scenario, the farmer is assumed to live on the site, raises crops and livestock onsite, and uses the groundwater for drinking water and irrigation. The pathways included in this assessment are:

- Direct exposure to external radiation from residual radioactivity,
- Internal dose from inhalation of airborne radionuclides,
- Internal dose from ingestion of:
    - Plants that are grown in media containing residual radioactivity and are irrigated with water containing residual radioactivity,

- Meat and milk from livestock that are fed with fodder grown in soil containing residual radioactivity and that drink water containing residual radioactivity,
- Drinking water containing residual radioactivity from a well,
- Fish from a pond containing residual radioactivity,
- Soil containing residual radioactivity.

3.5.3.2 Computer Code and Parameter Selection

The licensee used RESRAD Version 6.5 to perform site-specific dose modeling to generate the soil DCGL values. The parameter values used in RESRAD were classified as behavioral, metabolic, or physical. The behavioral and metabolic parameters were assigned deterministic values from NUREG/CR-5512 Volume 3, NUREG/CR-6697, or the RESRAD default library. Physical parameters that were based on site-specific data were also modeled using a deterministic value. The remaining physical parameters were prioritized based on their relevance in the dose calculations, the variability of the dose as a result of changes in the parameter value, the parameter type, and the availability of parameter-specific data. The highest priority parameters, Priority 1 and 2, were initially modeled stochastically using distributions from NUREG/CR-6697. Priority 3 parameters were modeled using deterministic values from NUREG/CR-5512 Volume 3, NUREG/CR-6697, or the RESRAD default library.

The licensee performed a sensitivity analysis to identify the parameters that most affect the calculated dose. Parameters were identified as sensitive if the partial rank correlation coefficient (PRCC) was greater than or equal to 0.25. The sensitive parameters were assigned the $25^{th}$ or $75^{th}$ percentile value of the distribution based on whether the parameter was positively or negatively correlated with dose, and the non-sensitive parameters were assigned a distribution or site-specific value or the median value of the distribution.

The area of the contaminated zone was assumed to be 30,000 $m^2$ based on the area of the largest contiguous Class 1 and Class 2 areas. The contaminated zone thickness was assumed to be 2.67 m. This thickness was based on the 75% of a site specific distribution. In the RAI responses for Chapter 6, PG&E performed an evaluation of the potential dose if contamination is found at depths deeper than 2.67 m. In this evaluation, the dose was assessed from a 1 m layer located from 2.67 m to 3.67 m below the ground surface. The evaluation showed that the potential dose from any residual contamination at depths below 2.67 m is small, so the calculated DCGL values are appropriate even if contamination is found at greater depths.

The properties of the soil in the contaminated and unsaturated zones were based on “clay loam” and the properties of the soil in the saturated zone were based on “silty clay loam”. These soil types were selected based on data from installation logs for groundwater monitoring wells on site. In the response to RAI 28, PG&E performed a sensitivity analysis to evaluate the effect of the soil type selected on dose. They found that modeling the soil as sand instead of clay did not significantly affect the projected dose because the dose was primarily due to water independent pathways.

3.5.3.3 Calculated Soil $DCGL_w$ Values

The soil $DCGL_w$ values calculated by PG&E are presented in the table below. These values were calculated by dividing the 25 mrem/yr dose criteria by the dose calculated using RESRAD for a unit radionuclide concentration (i.e., 1 pCi/g). These $DCGL_w$ values correspond to the concentration of the radionuclide in soil that is projected to result in a 25 mrem/yr dose. The

unity rule will be used to evaluate the dose contribution from multiple radionuclides as is described in Section 3.4.6 of this SER.

**Table 1 Soil $DCGL_w$ Values**

| Radionuclide | Soil $DCGL_w$ (pCi/g) |
|---|---|
| Am-241 | $2.50 \times 10^1$ |
| C-14 | 6.30 |
| Cm-243 | $2.90 \times 10^1$ |
| Cm-244 | $4.80 \times 10^1$ |
| Cm-245 | $1.70 \times 10^1$ |
| Cm-246 | $2.50 \times 10^1$ |
| Co-60 | 3.80 |
| Cs-137 | 7.90 |
| Eu-152 | $1.00 \times 10^1$ |
| Eu-154 | 9.40 |
| H-3 | $6.80 \times 10^2$ |
| Nb-94 | 7.10 |
| Ni-59 | $1.90 \times 10^3$ |
| Ni-63 | $7.20 \times 10^2$ |
| Np-237 | 1.10 |
| Pu-238 | $2.90 \times 10^1$ |
| Pu-239 | $2.60 \times 10^1$ |
| Pu-240 | $2.60 \times 10^1$ |
| Pu-241 | $8.60 \times 10^2$ |
| Sr-90 | 1.50 |
| Tc-99 | $1.20 \times 10^1$ |

3.5.3.4 Area Factors for Soil

Area factors were developed by PG&E for soil to assess areas of elevated contamination at the HBPP site. The area factors were developed using the same parameters as the soil $DCGL_w$ values, except for the area of the contaminated zone and the contaminated fractions of plant, meat, and milk food products. The contaminated fractions of plant, meat, and milk were adjusted for the smaller areas based on the equations from the RESRAD User's Manual. The calculated area factors are provided in the table below. Multiple hot spots in a survey unit will be evaluated as described in Section 3.4 of this SER.

**Table 2 Area Factors for Soil**

| | 2000 m$^2$ | 1000 m$^2$ | 500 m$^2$ | 100 m$^2$ | 50 m$^2$ | 10 m$^2$ | 5 m$^2$ | 1 m$^2$ |
|---|---|---|---|---|---|---|---|---|
| Am-241 | 1.0 | 1.0 | 2.0 | 8.7 | 16 | 49 | 77 | 190 |
| C-14 | 1.0 | 1.5 | 4.0 | 42 | 110 | 1000 | 2500 | 18000 |
| Cm-243 | 1.0 | 1.0 | 1.6 | 3.4 | 4.3 | 7.3 | 11 | 32 |
| Cm-244 | 1.0 | 1.0 | 2.0 | 9.7 | 19 | 76 | 120 | 280 |
| Cm-245 | 1.0 | 1.0 | 1.9 | 6.2 | 9.2 | 19 | 30 | 81 |
| Cm-246 | 1.0 | 1.0 | 2.0 | 9.7 | 19 | 76 | 130 | 280 |
| Co-60 | 1.0 | 1.0 | 1.1 | 1.3 | 1.4 | 2.2 | 3.3 | 10.0 |
| Cs-137 | 1.0 | 1.0 | 1.3 | 1.7 | 1.9 | 3.0 | 4.5 | 14.0 |
| Eu-152 | 1.0 | 1.0 | 1.0 | 1.1 | 1.3 | 2.0 | 3.0 | 9.1 |
| Eu-154 | 1.0 | 1.0 | 1.0 | 1.2 | 1.3 | 2.0 | 3.0 | 9.2 |
| H-3 | 1.0 | 1.1 | 2.1 | 10 | 21 | 100 | 200 | 930 |
| I-129 | 1.0 | 1.1 | 2.2 | 11 | 22 | 99 | 190 | 830 |
| Nb-94 | 1.0 | 1.0 | 1.0 | 1.2 | 1.3 | 2.0 | 3.0 | 9.0 |
| Ni-59 | 1.0 | 1.2 | 2.3 | 12 | 23 | 120 | 230 | 1200 |
| Ni-63 | 1.0 | 1.2 | 2.3 | 12 | 23 | 120 | 230 | 1200 |
| Np-237 | 1.0 | 1.0 | 2.0 | 9.0 | 16 | 56 | 98 | 350 |
| Pu-238 | 1.0 | 1.0 | 2.0 | 9.7 | 19 | 76 | 130 | 280 |
| Pu-239 | 1.0 | 1.0 | 2.0 | 9.7 | 19 | 76 | 130 | 290 |
| Pu-240 | 1.0 | 1.0 | 2.0 | 9.7 | 19 | 76 | 130 | 290 |
| Pu-241 | 1.0 | 1.0 | 2.0 | 8.8 | 16 | 49 | 78 | 190 |
| Sr-90 | 1.0 | 1.0 | 2.0 | 10 | 20 | 99 | 200 | 960 |
| Tc-99 | 1.0 | 1.0 | 2.0 | 10 | 20 | 100 | 200 | 1000 |

3.5.3.5 NRC Evaluation of Soil DCGL Values

NRC staff evaluated the proposed critical group, land use scenario, and set of pathways used in the assessment and concludes that they are acceptable and are consistent with NRC guidance. The use of a resident farmer scenario is conservative and is acceptable for this site as this site is expected to be used for industrial purposes and the dose from the resident farmer scenario bounds the dose for the industrial scenario.

The NRC staff evaluated the parameters and modeling performed to generate the $DCGL_w$ values and area factors for soil. As noted in RAI 36, the NRC staff believes that some aspects of the groundwater modeling in the probabilistic dose assessment may have resulted in an underestimation of the projected groundwater pathway dose. For example, parameters that are expected to be correlated are not (such as the $K_d$ values in the contaminated, unsaturated and saturated ones). Also, because parameters were not correlated, some realizations of the probabilistic analysis had combinations of parameters that are physically impossible, such as an effective porosity that is higher than the total porosity. However, analyses of the groundwater pathway dose by PG&E in RAI 28 and independent analyses of the groundwater pathway dose performed by the NRC staff conclude that the groundwater pathway dose does not contribute significantly to the overall dose. Therefore, the lack of correlation between groundwater pathway parameters in the probabilistic dose assessment is not expected to affect the overall projected dose. The NRC staff finds that the parameters selected and modeling performed for the water independent dose are appropriate for the HBPP site and the land use scenario

evaluated and are consistent with NRC guidance. The NRC staff also performed independent calculations of the $DCGL_w$ values and area factors for select risk-significant radionuclides and obtained comparable results as PG&E. Therefore, the NRC staff concludes that the dose modeling completed for the development of the $DCGL_w$ values and area factors for soil provides reasonable assurance that the use of these criteria will, at the time of license termination, result in a dose to the average member of the critical group that will not exceed the 25 mrem annual dose criterion in 10 CFR 20.1402.

### 3.5.4 Building Surface DCGL Values

DCGL values were derived by PG&E for building surfaces in buildings that will remain following license termination.

#### 3.5.4.1 Scenario and Exposure Pathways Evaluated

The building surface DCGL values were calculated based on the building occupancy scenario. This scenario is based on an adult who works in the buildings following decommissioning of the site. The pathways included in this assessment are:

- Direct exposure to external radiation from:
    - Material deposited on the walls, floor, and ceiling,
- Internal dose from inhalation of airborne radionuclides,
- Internal dose from inadvertent ingestion of radionuclides.

#### 3.5.4.2 Computer Code and Building Site Specific Parameters

RESRAD-BUILD Version 3.5 was used to generate the building surface DCGL values. The parameters values were assigned using a similar approach as was used in the development of the soil DCGL values. A sensitivity analysis was performed to identify the sensitive parameters. For the building occupancy scenario, a parameter was identified as sensitive if the absolute value of its PRCC was greater than or equal to 0.10 and non-sensitive if the absolute value of its PRCC was less than 0.10. The sensitive parameters were assigned the $25^{th}$ or $75^{th}$ percentile value of the distribution based on whether the parameter was positively or negatively correlated with dose.

In the calculation of the building surface DCGL values, residual contamination was assumed to exist on the walls, floor, and ceiling, and the receptor was assumed to be located in the center of the room. The receptor was assumed to remain in the room 45 hours per week for 52 weeks a year.

The building surface DCGL values were derived based on a room in the General Office Building that is the smallest room that will be continuously occupied. This room is 8.7 m long by 5.64 m wide by 2.49 m tall. The projected dose from contamination on building surfaces is larger for a smaller room, so the development of the DCGL values based on the geometry of this room is conservative.

The removable fraction of the residual contamination was assumed to be equal to 0.1. In the RAI responses for Chapter 6, PG&E stated that smears will be taken at each fixed-point location on building surfaces to verify that the removable fraction is 0.1 or less.

### 3.5.4.3 Calculated Building Surface $DCGL_w$ Values

The building surface $DCGL_w$ values calculated by PG&E are provided in the table below. These values were calculated by dividing the 25 mrem/yr dose criteria by the dose calculated using RESRAD-BUILD for a unit radionuclide concentration (i.e., 1 pCi/$m^2$) and converting the results to units of dpm/100 $cm^2$. These $DCGL_w$ values correspond to the concentration of the radionuclide on building surfaces that is projected to result in a 25 mrem/yr dose.

**Table 3 Building Surface $DCGL_w$ Values**

| Radionuclide | Building Surface DCGL (dpm/100 $cm^2$) |
|---|---|
| Am-241 | $3.00 \times 10^3$ |
| C-14 | $7.00 \times 10^6$ |
| Cm-243 | $4.30 \times 10^3$ |
| Cm-244 | $5.50 \times 10^3$ |
| Cm-245 | $2.20 \times 10^3$ |
| Cm-246 | $2.70 \times 10^3$ |
| Co-60 | $1.30 \times 10^4$ |
| Cs-137 | $4.60 \times 10^4$ |
| Eu-152 | $2.70 \times 10^4$ |
| Eu-154 | $2.50 \times 10^4$ |
| H-3 | $1.80 \times 10^8$ |
| Nb-94 | $1.90 \times 10^4$ |
| Ni-59 | $6.30 \times 10^7$ |
| Ni-63 | $2.40 \times 10^7$ |
| Np-237 | $2.40 \times 10^3$ |
| Pu-238 | $3.40 \times 10^3$ |
| Pu-239 | $3.10 \times 10^3$ |
| Pu-240 | $3.10 \times 10^3$ |
| Pu-241 | $1.40 \times 10^5$ |
| Sr-90 | $9.70 \times 10^4$ |
| Tc-99 | $9.60 \times 10^6$ |

Gross $DCGL_w$ values were generated for building surfaces based on the individual radionuclide $DCGL_w$ values and the relative ratios of the radionuclides measured in site samples. These samples were taken in drains and trenches inside buildings and on roofs. The samples with identifiable amounts of activity are from buildings that are currently designated for demolition. In these samples, Cs-137 contributed approximately 94% of the activity and Co-60 contributed approximately 6%. These fractions were used, along with the radionuclide specific $DCGL_w$ values, to obtain a gross beta/gamma $DCGL_w$ of $4.06 \times 10^4$ dpm/100 $cm^2$. There is limited data on the relative amounts of alpha emitters at HBPP, so a value of 3,000 dpm/100 $cm^2$ will be used for the gross alpha $DCGL_w$ based on the $DCGL_w$ for the most limiting prevalent alpha emitter, Am-241. The unity rule will be used to account for both the beta/gamma and alpha activity.

3.5.4.4 Area Factors for Building Surfaces

Area factors for building surfaces were derived based on activity on a single surface that is located at a distance of 1 m from the receptor. The other parameters and pathways are the same as those used in the derivation of the building surface DCGLs. As is described in the response to RAI 30, the most conservative area factor will be used to evaluate elevated areas if they exist.

**Table 4 Area Factors for Building Surfaces**

| | 100 $m^2$ | 50 $m^2$ | 10 $m^2$ | 8 $m^2$ | 6 $m^2$ | 5 $m^2$ | 4 $m^2$ | 3 $m^2$ | 2 $m^2$ | 1 $m^2$ |
|---|---|---|---|---|---|---|---|---|---|---|
| Am-241 | 1.0 | 2.0 | 9.9 | 12 | 16 | 20 | 25 | 33 | 49 | 97 |
| C-14 | 1.0 | 2.0 | 9.9 | 12 | 16 | 20 | 24 | 33 | 49 | 97 |
| Cm-243 | 1.0 | 2.0 | 9.5 | 12 | 16 | 19 | 23 | 30 | 45 | 89 |
| Cm-244 | 1.0 | 2.0 | 10 | 12 | 17 | 20 | 25 | 33 | 50 | 100 |
| Cm-245 | 1.0 | 1.8 | 5.9 | 6.9 | 8.6 | 9.9 | 12 | 15 | 21 | 39 |
| Cm-246 | 1.0 | 1.9 | 8.0 | 9.7 | 12 | 15 | 18 | 23 | 33 | 64 |
| Co-60 | 1.0 | 1.2 | 2.5 | 2.8 | 3.3 | 3.7 | 4.3 | 5.3 | 7.2 | 13 |
| Cs-137 | 1.0 | 1.3 | 2.8 | 3.2 | 3.8 | 4.2 | 4.9 | 6.0 | 8.2 | 15 |
| Eu-152 | 1.0 | 1.2 | 2.5 | 2.8 | 3.3 | 3.7 | 4.3 | 5.3 | 7.2 | 13 |
| Eu-154 | 1.0 | 1.2 | 2.5 | 2.8 | 3.3 | 3.7 | 4.3 | 5.3 | 7.2 | 13 |
| H-3 | 1.0 | 2.0 | 10 | 12 | 17 | 20 | 25 | 33 | 50 | 100 |
| I-129 | 1.0 | 1.9 | 8.0 | 9.7 | 13 | 15 | 18 | 23 | 34 | 65 |
| Nb-94 | 1.0 | 1.2 | 2.5 | 2.8 | 3.3 | 3.7 | 4.3 | 5.3 | 7.2 | 13 |
| Ni-59 | 1.0 | 2.0 | 10 | 13 | 17 | 20 | 25 | 33 | 50 | 100 |
| Ni-63 | 1.0 | 2.0 | 10 | 12 | 17 | 20 | 25 | 33 | 50 | 100 |
| Np-237 | 1.0 | 2.0 | 9.4 | 12 | 16 | 18 | 23 | 30 | 45 | 89 |
| Pu-238 | 1.0 | 2.0 | 10 | 13 | 17 | 20 | 25 | 33 | 50 | 100 |
| Pu-239 | 1.0 | 2.0 | 10 | 12 | 17 | 20 | 25 | 33 | 50 | 100 |
| Pu-240 | 1.0 | 2.0 | 10 | 12 | 17 | 20 | 25 | 33 | 50 | 100 |
| Pu-241 | 1.0 | 2.0 | 9.9 | 12 | 17 | 20 | 25 | 33 | 49 | 98 |
| Sr-90 | 1.0 | 2.0 | 9.5 | 12 | 16 | 19 | 23 | 30 | 45 | 90 |
| Tc-99 | 1.0 | 2.0 | 9.4 | 12 | 15 | 18 | 23 | 30 | 44 | 87 |

3.5.4.5 NRC Evaluation of Building Surface DCGL Values

The NRC staff evaluated the proposed critical group, scenario, and set of pathways used in the determination of the building surface DCGL values and finds that they are acceptable and are consistent with NRC guidance. The NRC staff also reviewed the parameter values assumed in the modeling and finds that they are appropriate for the site conditions and are consistent with NRC guidance. The NRC staff performed independent analyses using RESRAD-BUILD to confirm the building surface $DCGL_w$ values and area factors for risk-significant radionuclide. In this analysis, NRC staff obtained comparable results to PG&E.

Based on the analysis above, the NRC staff concludes that the dose modeling completed for the development of the $DCGL_w$ values and area factors for building surfaces provides reasonable assurance that the use of these criteria will, at the time of license termination, result in a dose to the average member of the critical group that will not exceed the 25 mrem annual dose criterion

in 10 CFR 20.1402 as long as the assumptions used in PG&E's analysis are found to be true. These assumptions include the removable fraction being less than 10%, the contamination not being volumetric, and the relative ratios of the individual radionuclides being consistent with those assumed in the development of the gross $DCGL_w$ values. These assumption will be verified as facility structures are removed.

3.5.5 Embedded and Buried Piping DCGL Value

A limited amount of embedded and buried piping is expected to remain after decommissioning at the HBPP site. Section 5.4.3.3 of the License Termination Plan states that all remaining embedded and buried piping will be grouted for stability after surveying unless it is to be used as an active system, such as drainage piping.

3.5.5.1 Development Embedded and Buried Piping DCGL Value

The development of a DCGL value for this piping is described in "Derived Concentration Guideline Levels for Embedded and Buried Piping in Support of the Final Status Survey at HBPP, Revision 1". A DCGL value of 100,000 dpm/100 $cm^2$ was selected based on this being a level that is readily detectable without requiring highly specialized instrumentation.

An assessment was performed by PG&E to evaluate the potential dose from piping with residual contamination of 100,000 dpm/100 $cm^2$. In this assessment, the piping was assumed to be completely degraded in year zero and the activity in the piping was assumed to be uniformly distributed over the volume of the piping. The piping was assumed to have a thickness of one meter and was assumed to have two meters of cover above it, and an area of 100 $m^2$ was assumed for the contaminated zone. This area is based on the assumption that all of the piping expected to remain at license termination is located in one location. This dose assessment was performed using a similar probabilistic calculation as was performed in the development of the soil DCGL values.

The radionuclide activities used in this analysis area based on the results of a sample taken in the off-gas tunnel. In the RAI responses, PG&E stated that this sample was selected because the beta and alpha activities were sufficiently present to allow for development of ratios from positive data and that buried piping to be surveyed will have analyses performed on scrapings/sediment present in the piping to assess that the ratios used are still representative.

The peak of the mean dose calculated in this assessment was $9.67 \times 10^{-4}$ mrem/y. PG&E stated in the RAI responses that the calculated dose contribution from the piping will be subtracted from the 25 mrem/yr dose limit for survey units that lay atop the piping.

3.5.5.2 NRC Evaluation of Embedded and Buried Piping DCGL Value

The NRC staff finds that although there are differences between the assumed conceptual model and expected configuration of residual contamination in the piping, the assumed conceptual model for the embedded and buried piping is acceptable for the following reasons: 1) the assumptions that the piping will be completely degraded at the time of license termination and the piping is all located in the same place are conservative because both of these assumptions result in a higher projected dose; and 2) the assumption that the material along the edges of the pipe is evenly distributed over the volume of the pipe could be non-conservative if the external exposure pathway dose was significant, however, due to the amount of cover over the pipes,

the primary contribution to dose is through the water dependent pathways, so the configuration of the contamination within the piping itself is not expected to affect the projected dose.

The proposed method of subtracting the dose from the piping from the total 25 mrem/yr dose limit for survey units that lay atop the piping is an acceptable method of accounting for the dose from both the piping and the soil in the survey unit.

Based on the analysis above, the NRC staff concludes that the $DCGL_w$ value proposed for the buried and embedded piping provides reasonable assurance that the 25 mrem annual dose criterion in 10 CFR 20.1402 will not be exceeded at the time of license termination. These assumptions include the piping being at a depth of greater than 2 m and the relative ratios of radionuclides being consistent with those assumed in the dose calculation for the piping.

### 3.5.6 Conclusion

The NRC staff has reviewed the dose modeling analyses used to generate DCGL values for soil, building surfaces, and buried and embedded piping at the HBPP Unit 3 site as part of the review of the LTP, using NUREG-1757, Volume 2. Based upon the analyses above, the staff concludes that the dose modeling is reasonable and is appropriate for the exposure scenarios under consideration. The NRC staff concludes that the DCGL values developed for soil, building surfaces, and buried and embedded piping at the HBPP site are consistent with the 25 mrem/yr annual dose criterion for unrestricted release in 10 CFR 20.1402. This conclusion is based on the modeling effort performed by PG&E and the independent, confirmatory analyses performed by the NRC staff.

## 3.6 Groundwater

The NRC staff has evaluated the geologic and hydrogeologic conditions at the HBPP to determine the extent of residual radioactive contamination in the groundwater. The evaluation and summary of information below is based upon information provided in the HBPP LTP, supporting documents referenced in the LTP, and NRCs independent assessment of the LTP and supporting documents.

### 3.6.1 Geology

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 914 to 1,219 m [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.

### 3.6.2 Stratigraphy

The main stratigraphic formation of interest underlying HBPP is the Pleistocene Hookton Formation that is approximately 335 m [1,100 ft] thick. The Pleistocene marine terrace deposits typically overlie the Hookton Formation and are generally included as part of the formation. However, the Hookton Formation is locally overlain by Holocene Bay Deposits of Humboldt Bay and by Holocene alluvial deposits along the streams in the region. The Hookton Formation

contains several important groundwater aquifers both locally and regionally. The Hookton Formation unconformably overlies the Pleistocene Scotia Bluffs Formation.

### 3.6.3 Hookton Formation

The Hookton Formation consists of interbedded shallow-water marine, estuarine, and fluvial deposits of sand, silty sand, chert-rich gravel, and clay. The formation is divided into an upper and lower unit. The upper unit is approximately 18 to 24 m [60 to 80 ft] thick and consists of laterally discontinuous beds of clay and silt, and sand and gravel that change laterally with interfingering, cut-and-fill, and gradational facies changes. The clay beds consisting of ancient bay sediments are laterally persistent compared to the interbedded sandy and silty layers. The Hookton Formation near the Buhne Point Hill has been tectonically tilted to the east a few degrees toward the intake and discharge canals. The Discharge Canal fault has displaced the Hookton Formation with the south side being up-thrown compared to the north side.

### 3.6.4 Upper Hookton Formation

The upper Hookton Formation can be divided into two informal lithologic units that are referred to as the upper Hookton silt and clay beds and the upper Hookton sand beds. The upper Hookton silt and clay beds consist of silt, clay, and silty sand beds that extend from the surface to a depth of approximately 4.5 to 10.5 m [15 to 35 ft]. The lower portion consists of clay and silt beds referred to as the first bay clay.

Underlying the upper Hookton silt and clay beds are the upper Hookton sand beds. The sand beds are approximately 8 to 12 m [25 to 40 ft] thick and consist of sand and gravel layers with lesser silt and clay beds.

The upper Hookton sand beds overlie a discontinuous clay bed that is referred to as the second bay clay. The second bay clay is discontinuous across the site, but underlies a portion of HBPP Unit 3 to the south and east where it is approximately 2.5 to 4 m [8 to 13 ft] thick.

### 3.6.5 Lower Hookton Formation

The lower Hookton Formation consists of laterally persistent beds of alternating sand, silty sand, gravel, gravely sand, silty clay, and clay. The upper 8 to 46 m [26 ft to 150 ft] consists of sand and gravel that overlies the Unit F clay. The Unit F clay, which is approximately 15 m [50 ft] thick, is a distinctive marker bed with relatively low permeability that functions as a regional aquitard. Beneath the Unit F clay are alternating layers of clean, well-sorted sand and clay that extends to a depth of approximately 335 m [1,100 ft].

### 3.6.6 Aquifers

The aquifers of importance in the vicinity of HBPP Unit 3 are contained in the Hookton Formation and are referred to as the zone of perched groundwater in the upper Hookton silt and clay beds, the Upper Hookton aquifer, and the aquifer between Unit F and the second bay clay. The perched groundwater zone is located above the first bay clay unit. The Upper Hookton aquifer lies between the first bay clay unit and the second bay clay unit where present. The second bay clay unit is not present to the west and north of the HBPP. The Upper Hookton sand beds were found to transition directly into the texturally similar deposits of the Lower Hookton Formation where the second bay clay was not present. The Upper Hookton aquifer and the aquifer between Unit F and the second bay clay unit will be referred to as the Upper

Hookton aquifer due to the similarities and hydraulic connection of the two aquifers downgradient of the HBPP. Below the Unit F clay is the Lower Hookton aquifer. The Lower Hookton aquifer is considered to be a fresh water bearing zone.

### 3.6.7 Zone of Perched Groundwater in the Upper Hookton Silt and Clay Beds

Discontinuous zones of perched groundwater are encountered between the ground surface and the first bay clay within the Upper Hookton Formation. The discontinuous nature of the water bearing deposits limits lateral flow. The groundwater flow direction of the perched water is generally to the north where present. The groundwater within the upper silt and clay beds does not appear to be tidally influenced.

### 3.6.8 Upper Hookton Aquifer

Above the Unit F clay aquitard and below the first bay clay is the shallow, brackish-water aquifer referred to as the upper Hookton aquifer. The upper Hookton aquifer consists of both the Upper Hookton Formation and a section of the Lower Hookton Formation above the Unit F clay. The upper Hookton aquifer is over 33.5 m [110 ft] thick and is confined by the first bay clay aquitard in the vicinity of HBPP unit 3. The upper Hookton aquifer is locally considered a semi-confined aquifer due to the unconfined conditions to the west of HBPP Unit 3 near Buhne Point Hill. The aquifer is comprised of sand and gravel lenses, including some clean sand strata. The second bay clay, where present, is approximately 6 to 9 m [20 to 30 ft] below the bottom of the first bay clay. The second bay clay varies in thickness and is discontinuous north and west of HBPP Unit 3. The piezometric surface of the upper Hookton aquifer generally slopes north toward Humboldt Bay. The piezometric surface north of the Discharge Canal fault generally slopes toward the northwest with higher water levels on the north side of the fault than the south side indicating the fault is acting as an aquitard.

The tides have a strong influence on the upper Hookton piezometric surfaces. The piezometric surface lags the tidal changes by a few hours with up to approximately 1 m [3 ft] of elevation change during a tidal cycle. The tidal fluctuations produce significant short-term (hours) effects on the groundwater flow directions and rates within the upper Hookton aquifer. During rising tides, bay water flows into the formation in a generally southerly direction; during falling tides, the flow is out of the formation into the bay, generally in a northerly direction. However, the upper Hookton aquifer is believed to have a net discharge of groundwater into Humboldt Bay.

### 3.6.9 Lower Hookton Aquifer

The lower Hookton aquifer lies below the 15 m [50 ft] thick regional aquitard known as the Unit F clay. The lower Hookton aquifer is defined as the freshwater bearing zone of clean, sorted sands underling the Unit F clay at a depth of approximately 72 m [235 ft] bgs in the vicinity of HBPP Unit 3. The groundwater in the lower Hookton aquifer is typically utilized in wells above 137 m [450 ft] in depth even though the sand layers extend to greater depths. This confined aquifer is artesian in places. Recharge to the lower Hookton aquifer is believed to be through deep percolation of rainfall into formation outcrops beneath the terraces of Humboldt Hill and from alluvium along the Elk River Valley. Lateral flow in the aquifer transports water to area of discharge in Humboldt Bay and probably into the Pacific Ocean beyond Humboldt Bay.

### 3.6.10 Liquid Radiological Spills, Leaks, and Releases

The licensee has reported the occurrences of spills, leaks, and releases of site generated radionuclides to the environment in Table 2-2 of Section 2.1.6 to the LTP. The primary areas of concern regarding impacts to groundwater are generally near the SFP, the reactor caisson, and the Radwaste treatment building.

Of particular concern was the March 1966 discovery that a leak in the SFP liner had developed and approximately 55 gallons per day was draining from the SFP. Outflow of water into the surrounding environment was minimized by maintaining the water level in the gap between the liner and SFP. The result was an influx of groundwater at a rate of approximately 12 gallons per day.

Investigations were conducted to determine the magnitude of any groundwater contamination that could have occurred. Samples from the two plant domestic water wells and the reactor caisson sump did not reveal signs of contamination. About early April 1966, a sample well was installed north of the SFP and it revealed evidence of contamination, but the levels were a factor of 100 below allowable drinking water limits. Two additional wells were installed in May of 1966 to the east and south of the spent fuel storage building. Neither well showed any signs of contamination. Samples of water and sediment from the SFP liner and from the French drain showed Zn-65 levels comparable to the SFP water.

In the period from 1981 to 1984, concentrations of Cs-137 and Co-60 from the French drain were evaluated and shown to have similar ratios of these radionuclides to samples from the liner gap and the SFP. The significantly lower concentration of these radionuclides from water collected from the liner gap suggested that infiltrating groundwater was the major contributor of water to the liner gap. Analysis of the volumes of water and the concentrations indicated that the SFP inflow rate has generally been less than 2 gallons per day since 1990.

### 3.6.11 Groundwater Monitoring

Assessments of any residual activity in groundwater at the HBPP will be through groundwater monitoring wells. 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 for the resident farmer (i.e., resident farmer's well) in chapter 6 of the LTP, 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 10 CFR 20.1402.

The number and type of groundwater monitoring wells were historically required by the SAFSTOR Offsite Dose Calculation Manual (ODCM), which contained the requirements for the Radiological Environmental Monitoring Program (REMP).

The REMP required groundwater monitoring at intermediate and deep screened wells. The intermediate screened groundwater monitoring wells consisted of MW-1, MW-2, MW-4, MW-6, MW-11, RCW-SFP-1, and RCW-SFP-2. The deep screened groundwater monitoring wells consisted of RCW-CS-1, RCW-CS-2, RCW-CS-3, RCW-CS-4, and RCW-CS-5. The seven intermediate groundwater monitoring wells contain screened intervals at approximately 35 to 67 ft below ground surface (bgs) in the upper part of the Hookton aquifer. The five deep groundwater monitoring wells contain screened intervals at approximately 73 to 94 ft bgs in the lower Hookton aquifer. The twelve monitoring wells were sampled quarterly to satisfy the

groundwater monitoring requirements within the vicinity of HBPP. Table 2-20 of the LTP lists the monitoring wells and provides monitoring well elevation and depth.

Additional shallow groundwater monitoring wells were installed in 2009. The wells were screened at approximately 15 to 25 ft bgs within the 1st bay clay unit. Groundwater monitoring at the shallow wells was not required under the REMP, but several of these wells have been sampled and groundwater measurements were taken during quarterly monitoring events.

In order to accommodate necessary changes to site access and for the removal of soils and building structures during decommissioning, the groundwater monitoring requirements were transferred from the ODCM to HBPPs groundwater monitoring program procedures. Performing groundwater monitoring through procedure rather than the ODCM provides the licensee with the necessary flexibility to adjust the number and location of groundwater monitoring wells throughout decommissioning. Additional wells will be installed for future monitoring to evaluate the impact of decommissioning activities on groundwater.

### 3.6.12 Groundwater Sampling and Analysis for Radionuclides of Concern

Historically, groundwater monitoring wells were sampled quarterly and samples were analyzed for gamma emitting isotopes, gross alpha and beta, and tritium. The high salinity of the groundwater makes the achievement of the required minimum detectable activities (MDAs) problematic for gross alpha and beta radioactivity. The MDA is dependent on the detection capability for a given instrument, procedure, and the type of sample. Gross alpha and beta analyses for monitoring potential SFP leakage to the groundwater was determined to be less effective than tritium and gamma emitting isotopes. Analyses for gross alpha and beta are performed as a matter of plant policy, but this is not a formal regulatory requirement. The inability to achieve the required MDAs, due to the salinity of the groundwater samples, resulted in additional analyses for isotopic Am-241 and isotopic Sr-90 starting in the first quarter of 2006.

Groundwater leakage into the reactor caisson sump and from the french drain below the SFP are also routinely sampled and analyzed for gamma emitters and tritium. Detectable concentrations have been reported from these monitoring locations, but remain well below the EPA MCLs for drinking water.

Detectible concentrations of radionuclides have been reported within the groundwater of the HBPP Unit 3 restricted area. The detectable concentrations observed are likely the result of historical spills. However, the detectable concentrations for radiological constituents have not exceeded EPA MCLs for drinking water.

### 3.6.13 Influences to Hydrologic Condition during Decommissioning

Installation of a slurry wall around the HBPP Unit 3 reactor caisson and SFP is currently being done to facilitate removal of the structures and provide a barrier for inflowing groundwater during excavation and dewatering activities. The slurry wall will extend to the Unit F clay at a depth of approximately 185 ft bgs. Dewatering wells or well points will be installed after the slurry wall is completed to facilitate removal of the groundwater to a level beneath any active excavations. The dewatering system will require continuous pumping to maintain the necessary groundwater levels. Groundwater outside of the slurry wall will be forced to flow around the structure, which will alter the localized groundwater flow regime. The barrel-shaped slurry wall is generally parallel to the groundwater flow direction and will not cut off or inhibit local groundwater movement. Groundwater may mound on the up-gradient side of the slurry wall

and a stagnation area or depression may form on the down-gradient side. Groundwater flow velocity may increase around the lateral margins of the structure where the groundwater gradient would be highest. However, any changes to groundwater flow are expected to be minimal due the tidal influence from Humboldt Bay on the groundwater flow direction.

### 3.6.14 Groundwater Treatment

Groundwater collected during the installation of the slurry wall and removal of subsurface structures will be treated and released using the Ground Water Treatment System (GWTS) in accordance with the Storm Water Pollution Prevention Plan (SWPP) and the associated National Pollutant Discharge Elimination System (NPDES) permit. The GWTS is an Active Treatment System (ATS) designed to remove suspended solids in order to meet release criteria of the SWPP. The system will be limited to treating water containing soluble radionuclides less than 10 times the 10 CFR 20, Appendix B, Table 2, Column 2 effluent concentration limits (ECLs) in order to ensure that the instantaneous concentration of radioactive material released beyond the Site Boundary will be less than or equal to 10 times the concentrations specified in 10 CFR Part 20, Appendix B, Table 2, Column 2. The treatment system will have a maximum capacity of approximately 1,136 Liters per minute (300 gallons per minute).

### 3.6.15 Conclusion

The staff finds that the geologic and hydrogeologic site characterization provided in the LTP and referenced reports is sufficiently detailed and fulfills the site characterization requirement of 10 CFR 50.82(a)(9)(ii)(A). The information provided will allow the NRC staff to determine the extent and range of radiological contamination to the groundwater prior to license termination. While detectable concentrations of radionuclides have been reported within the groundwater of the HBPP Unit 3 restricted area, concentrations for radiological constituents have not exceeded EPA MCLs for drinking water. No imminent threats to human health or the environment due to radiological constituents in the groundwater have been identified, although additional assessments, i.e., groundwater monitoring, during decommissioning will be ongoing. In addition, the groundwater is saline and is not currently or likely to be used in the future for direct consumption or for agricultural purposes.

The LTP has identified radiological spills, leaks, and releases with the potential of impacting groundwater. The identified spills, leaks, and releases have not resulted in impacts to the groundwater based on the analyses performed on samples obtained from groundwater monitoring wells. Additionally, the extensive subsurface work proposed to remove below grade structures will require extensive dewatering of the aquifers. The dewatering activities will essentially act as a typical groundwater pump and treat system even though impacts from residual radioactive contamination in the groundwater have not been detected.

The NRC will obtain split samples from the groundwater monitoring wells prior to license termination. The samples will be analyzed for all of the primary constituents of concern identified in the LTP.

## 3.7 Site Specific Cost Estimate

### 3.7.1 Regulatory Requirements and Criteria

In accordance with 10 CFR 50.82(a)(9)(ii)(F), the LTP must include an updated site-specific estimate of remaining decommissioning costs.

### 3.7.2 Evaluation of the Updated Site-Specific Estimate of Remaining Decommissioning Costs

In Chapter 7 of the LTP, PG&E estimated the remaining decommissioning costs associated with the termination of the HBPP Unit 3 to be $727.6 million (2011 dollars). This is consistent with the cost estimate submitted to and approved by the California Public Utilities Commission (CPUC) for the HBPP Unit 3 2012 Nuclear Decommissioning Cost Triennial Proceeding (NDCTP). According to Revision 1 of the LTP, with system dismantling work underway, PG&E did not update the previous cost studies performed by TLG Services, Inc.; rather the current estimate reflects forecasts that have been developed from engineering studies and/or actual contractor bids. The estimate update incorporates the site specific and special tasks that have been prescribed or implemented as a result of the ongoing decommissioning planning. The basis of the estimate and the sources of information, methodology, site-specific considerations, assumptions, and total costs were presented in the revision to the LTP.

Prior to starting the detailed review of the cost estimate, the NRC staff reviewed the estimate to confirm the supporting systems/structures necessary to support the safe operation had been identified in the estimate. The validity of the cost estimate is based on a reasonable estimate of the cost to decommission the remaining supporting systems and structures, as well as confirming that all of the major equipment necessary to support operation was included.

In Revision 1 to the LTP, the licensee has divided the estimated remaining decommissioning costs (2011 dollars) into the following Cost Categories: General Staffing (Excludes Caisson); Remainder of Plant Systems; Site Infrastructure; Specific Project Costs (Excludes Disposal/Caisson/Canal); Waste Disposal (Excludes Caisson/Canals); Small Value Contracts; Contingency (Excludes Caisson/Canals); Caisson; Canal Remediation; and Common Site Support – Caisson and Canals. The estimated total remaining cost of decommissioning based on the above factors, as of December 2013, is $665.02 million. This estimate includes overall average contingency of 18.2 percent across the major activities. The staff reviewed the contingency factors and the work difficulty factors used in the PG&E cost estimate and found them to be reasonable.

### 3.7.3 Evaluation of the Decommissioning Funding Plan

According to Revision 1 of the LTP, PG&E filed an application with the CPUC requesting approval of the 2012 NDTCP revised cost estimates and authorization to recover the increased decommissioning funds in the nuclear decommissioning charge paid by PG&Es customers. The CPUC subsequently approved the cost estimate and issued a decision authorizing $680 million (corrected) to complete the decommissioning of HBPP Unit 3. The staff noted that this amount is 93 percent of PG&E's request for funds to complete decommissioning. The amount not authorized in the 2012 NDTCP decision, as well as any new justifiable costs, may be presented during later triennial cost proceedings.

In addition to the requirement to provide an updated site-specific estimate of remaining decommissioning costs for the evaluation of the LTP, PG&E is also required to provide annual decommissioning funding status reports for HBPP Unit 3. Pursuant to 10 CFR 50.82(a)(8)(v), the licensee is required to provide an annual financial assurance status report, by March 31 of each year. The annual report includes information related to decommissioning costs, expenditures, and funds. Accordingly, on March 31, 2015, PG&E submitted 2014 HBPP Unit 3 Decommissioning Funding Status (DFS) Report (ADAMS ML15090A774). The subsequent NRC staff analysis of the annual DFS report was based on a decommissioning trust fund (DTF) balance for radiological decommissioning of $191.5 million as of December 31, 2013. Staff

applied a real rate of return of 2.0 percent to its analysis through the expected license termination year of 2025 and credited additional CPUC-authorized collections of $96 million per year for years 2014 – 2017, totaling $385 million, resulting in a surplus of funds over the estimated $665 million needed to complete decommissioning.

The NRC staff finds the site-specific cost estimate for remaining radiological decommissioning costs for HBPP Unit 3 is reasonable, and that the DTF balance, as of December 31, 2013, will be sufficient to fund the remaining radiological decommissioning expenses.

#### 3.7.4 Conclusion

The NRC staff finds that decommissioning cost estimate and decommissioning funding plan associated with PG&E's License Termination Plan for HBPP Unit 3 are adequate and provide sufficient details associated with the funding mechanisms. The NRC staff, therefore, concludes that the licensee's LAR, "Addition of License Condition 2.C.5, 'License Termination Plan,'" for HBPP Unit 3 complies with 10 CFR 50.82(a)(9)(ii)(F).

### 3.8 Environmental Report

In accordance with the requirements of 10 CFR 50.82(a)(9)(ii)(G), the licensee is required to provide a supplement to the environmental report, pursuant to 10 CFR 51.53, describing any new information or significant environmental changes associated with the licensee's proposed license termination activities. Section 8 of the LTP updates the environmental information provided previously by the licensee both pre and post-operation. Therefore, Section 8 of the LTP constitutes a supplement to the HBPP Environmental Report, as required by 10 CFR 51.53(d) and 10 CFR 50.82(a)(9)(ii)(G). Based on the information in Section 8, the licensee concluded that the environmental impacts associated with changes in the licensee's decommissioning activities for HBPP Unit 3 remain bounded by the previously issued "Final Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities," NUREG-0586. Under the provisions of 10 CFR 51.21, the staff prepared an environmental assessment (EA) to determine the impacts of the proposed action on the environment. Based upon this EA, the staff found that approval of the LTP would not cause any significant impacts on the human environment and is protective of human health; the staff has prepared a Finding of No Significant Impact to formalize this finding.

The staff has reviewed the information in the LTP for HBPP, according to Section B.8 of NUREG-1700. Based on this review and the EA prepared by the staff, the staff has determined that the licensee has met the requirements of 10 CFR 50.82(a)(9)(ii)(G) and 10 CFR 51.53.

### 3.9 Change Procedure

The licensee has proposed that it be authorized to make certain changes to the NRC-approved LTP without NRC approval if these changes do not: (1) require NRC approval pursuant to 10 CFR 50.59; (2) violate the requirements of 10 CFR 50.82(a)(6); (3) increase the probability of making a Type I decision error above the level stated in the LTP; (4) increase the radionuclide specific DCGLs and related minimum detectable concentrations; (5) increase the radioactivity level, relative to the applicable DCGL, at which investigation occurs; or (6) change the statistical test applied to one other than the Sign Test or Wilcoxon Rank Sum Test. If the licensee elects to reduce a survey unit's classification (i.e., from Class 1 to Class 2 or 3, or from Class 2 to 3), prior notification will be provided to NRC at least 14 days prior to implementation. The licensee will submit changes to the LTP not requiring NRC approval as an update to the final safety

analysis report, in accordance with 10 CFR 50.71(e). The staff concludes that authorizing the licensee to make certain changes, during the final site remediation, is acceptable, subject to the above listed conditions.

## 4.0 STATE CONSULTATION

In accordance with the NRC's regulations, Stephen Hsu, Radiological Health Branch of the California Department of Public Health, was notified of the proposed issuance of the amendment. The State official had no comments.

## 5.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an EA and Finding of No Significant Impact were prepared and published in the Federal Register on May 3, 2016.

Based on the EA, the NRC has determined that issuance of this amendment will not have a significant effect on the quality of the human environment. Accordingly, the NRC has determined that a Finding of No Significant Impact is appropriate.

## 6.0 CONCLUSIONS

The NRC has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner; (2) such activities will be conducted in compliance with the NRC's regulations; and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

## 7.0 LIST OF CONTRIBUTORS

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## 8.0 LIST OF ACRONYMS

| | |
|---|---|
| AEC | Atomic Energy Commission |
| AF | Area Factor |
| ALARA | As Low As Is Reasonable Achievable |
| ANL | Argonne National Laboratory |
| bgs | below ground surface |
| BMP | Best Management Practice |
| Bq/g | Becquerel per gram |
| Bq/L | Becquerel per liter |
| CAB | Citizens Advisory Board. |
| CCC | California Coastal Commission |
| CCS | Continuing Characterization Survey |
| CDP | Coastal Development Permit |

| | |
|---|---|
| CEC | California Energy Commission |
| CFR | Code of Federal Regulations |
| CFS | Containment Foundation Sump |
| CIRP | Caisson In Leakage Repair Project |
| cm | centimeters |
| $cm^2$ | square centimeter |
| COC | Chain of Custody |
| CPUC | California Public Utilities Commission |
| CRB | Count room building |
| CWT | Concentrated Waste Tank |
| D&D | Decontamination & Decommissioning |
| DAW | Dry Activated Waste |
| DCF | Dose Conversion Factor |
| DCGL | Derived Concentration Guideline Limit |
| $DCGL_{EMC}$ | DCGL that represents the same dose to an individual for residual radioactivity in a smaller area within a survey unit. |
| $DCGL_W$ | DCGL for the average residual radioactivity in a survey unit |
| DOE | U.S. Department of Energy |
| DP | Decommissioning plan |
| dpm | disintegrations per minute |
| dpm/100$cm^2$ | disintegrations per minute per 100 square centimeters |
| DPR | Decommissioning Project Report |
| DQA | Data Quality Assessment |
| DQO | Data Quality Objective |
| DSAR | Defueled Safety Analysis Report |
| DTSC | Department of Toxic Substances Control |
| EA | Environmental Assessment |
| EMC | Elevated Measurement Comparison |
| EPA | U.S. Environment Protection Agency |
| ETD | Easy to detect |
| FR | Federal Register |
| $Ft^3$ | cubic foot |
| FGR | Federal Guidance Report |
| FSME | Office of Federal and State Materials and Environmental Management Programs |
| FSS | Final Status Survey |
| GEIS | Generic Environmental Impact Statement |
| GTCC | Greater than Class C |
| HBGS | Humboldt Bay Generating Station |
| HBPP | Humboldt Bay Power Plant |
| HEPA | High Efficiency Particulate Air filter |
| HPGe | High Purity Germanium |
| HSA | Historical Site Assessment |
| HSE | Health, Safety, and Environment |
| HTD | Hard to Detect |
| IA | Industrial Area |
| ICS | Initial Characterization Survey |
| ISFSI | Independent Spent Fuel Storage Installation |
| ISOCS | In Situ Object Counting System |
| kV | kilovolt |
| LBGR | Lower Boundary of the Gray Region |
| LLRW | Low-level Radioactive Waste |

| | |
|---|---|
| LPG | Liquid Propane Gas |
| LTP | License Termination Plan |
| $m^2$ | square meter |
| $m^3$ | cubic meter |
| MARSSIM | Multi-Agency Radiation Survey And Site Investigation Manual |
| MDC | Minimum Detectable Concentration |
| MeV | Mega electron Volts |
| uR/hr | microroentgen per hour |
| mrem/hr | millirem per hour |
| mrem/yr | millirem per year |
| MSL | Mean Sea Level |
| mSv/yr | milliSievert per year |
| nC/Kg-hr | nanocoulomb per kilogram per hour |
| NEI | Nuclear Energy Institute |
| NIST | National Institute of Standards and Technology |
| NRC | Nuclear Regulatory Commission |
| OWS | Oil/Water Separator |
| PAB | Primary Auxiliary Building |
| PCB | Polychlorinated Biphenyl |
| pCi/g | picocurie per gram |
| pCi/L | picocurie per Liter |
| PG&E | Pacific Gas and Electric |
| $PM_{10}$ | particular matter of 10 microns |
| PRCC | partial rank correlation coefficient |
| PSDAR | Post-Shutdown Decommissioning Activities Report |
| QA | Quality Assurance |
| QAPP | Quality Assurance Project Plan |
| QC | Quality Control |
| RA | Restricted Area |
| RAI | Request for Additional Information |
| RCA | Radiologically Controlled Area |
| RCRA | Resource Conservation and Recovery Act |
| REMP | Radiological Environmental Monitoring Program |
| RWP | Radiation Work Permit |
| RWST | Refueling Water Storage Tank |
| $S_o$ | Sensitivity Threshold |
| SAFSTOR | A method of decommissioning in which a nuclear facility is placed and maintained in a condition that allows the facility to be safely stored and subsequently decontaminated (deferred decontamination) to levels that permit release for unrestricted use. |
| SCC | Secondary Component Cooling |
| SCM | Site Conceptual Model |
| SER | Safety Evaluation Report |
| SFP | Spent Fuel Pool |
| SRP | Standard Review Plan |
| Sv/hr | Sievert per hour |
| TBD | Technical Basis Documents |
| TCP | Traffic Control Plan |
| TEDE | Total Effective Dose Equivalent |
| TRU | Transuranic |
| VSP | Visual Sample Plan software |

WMB Waste management building
WWI Wastewater Impoundments

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