November 27, 2007

Mr. Steve J. Redeker, Manager
Plant Closure & Decommissioning
Sacramento Municipal Utility District
14440 Twin Cities Road
Herald, CA 95638-9799

SUBJECT: RANCHO SECO NUCLEAR GENERATING STATION - ISSUANCE OF AMENDMENT RE: LICENSE TERMINATION PLAN (TAC NO. J52668)

Dear Mr. Redeker:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 133 to Facility Operating License (Possession Only) No. DPR-54 for the Rancho Seco Nuclear Generating Station. The amendment consists of changes to the license in response to your application dated April 7, 2005, April 12, 2006, August 3, 2006, and May 31, 2007.

The amendment revises the Rancho Seco Nuclear Generating Station License to add License Condition 2.C.(4) to the Rancho Seco license. This new license condition incorporates the, NRC approved, "License Termination Plan" (LTP), and associated addendum, into the Rancho Seco 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 NRC's next biweekly *Federal Register* notice.

Sincerely,

/RA/

John B. Hickman, Project Manager
Reactor Decommissioning Section
Decommissioning and Uranium Recovery
Licensing Directorate
Division of Waste Management
and Environmental Protection
Office of Federal and State Materials and
Environmental Management Programs

Docket No. 50-312

Enclosures:
1. Amendment No. 133 to DPR-54
2. Safety Evaluation

cc: Rancho Seco Service List

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Plant Closure & Decommissioning
Sacramento Municipal Utility District
14440 Twin Cities Road
Herald, CA 95638-9799

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DATE	10/16/2007	10/10/2007	10/12/ 2007	10/16/ 2007	10/25/2007	10/21/2007	11/26/2007

OFFICIAL RECORD COPY

Rancho Seco Nuclear Generating Station Service List

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SACRAMENTO MUNICIPAL UTILITY DISTRICT

DOCKET NO. 50-312

RANCHO SECO NUCLEAR GENERATING STATION

AMENDMENT TO FACILITY OPERATING LICENSE (POSSESSION ONLY)

Amendment No. 133
License No. DPR-54

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

   A. The application for amendment filed by the Sacramento Municipal Utility District (the licensee) dated April 7, 2005, April 12, 2006, August 3, 2006, and May 31, 2007, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the rules and regulations of the Commission;

   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.(4) is added to read as follows:

   (4) License Termination Plan (LTP)

   NRC License Amendment No. 133 approves the License Termination Plan.

   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 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.

Re-classification 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, re-classification to a less restrictive classification (e.g., 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 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Keith I. McConnell, Deputy Director
Decommissioning and Uranium Recovery
Licensing Directorate
Division of Waste Management
and Environmental Protection
Office of Federal and State Materials and
Environmental Management Programs

Date of Issuance: November 26, 2007

ATTACHMENT TO LICENSE AMENDMENT NO. 133

FACILITY OPERATING LICENSE (POSSESSION ONLY) NO. DPR-54

DOCKET NO. 50-312

Replace the following pages of the license with the attached revised pages. The revised pages contain margin lines indicating the areas of change and amendment number and date.

REMOVE	INSERT
4	4
5	5

(3) Confirmatory Order — Amend. # 132 9/21/05

The movement of nuclear fuel into the Reactor Building in prohibited without prior NRC approval.

(4) License Termination Plan (LTP) — Amend. # 133 11/26/07

NRC License Amendment No. 133 approves the License Termination Plan.

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 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

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

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D. This license is subject to the following additional condition for the protection of the environment:

If harmful effects or evidence of irreversible damage are detected by the monitoring programs including in the Rancho Seco Quality Manual, the Applicant will provide an analysis of the problem and a proposed course of action to alleviate the problem. — Amend. # 132 9/21/05

E. This license is effective as of the date of issuance and shall expire at midnight, October 11, 2008.

FOR THE ATOMIC ENERGY COMMISSION

/s/ Roger S. Boyd for

A. Giambusso, Deputy Director
for Reactor Projects
Directorate of Licensing

Attachment:
Appendix A - Technical Specifications

Amend. # 120 10/13/92

Date of Issuance:
August 16, 1974

SAFETY EVALUATION BY OFFICE OF FEDERAL AND STATE MATERIALS

AND ENVIRONMENTAL MANAGEMENT PROGRAMS

RELATED TO AMENDMENT NO. 133

TO FACILITY OPERATING LICENSE (POSSESSION ONLY) NO. DPR-54

SACRAMENT MUNICIPAL UTILITY DISTRICT

RANCHO SECO NUCLEAR GENERATING STATION

DOCKET NO. 50-312

1.0 INTRODUCTION

By letters dated April 7, 2005[1], April 12, 2006[2], August 3, 2006[3], and May 31, 2007[4], Sacramento Municipal Utility District (SMUD or the licensee) submitted a request to amend Facility Operating License No. DPR-54 for the Rancho Seco Nuclear Generating Station (Rancho Seco or the facility). In accordance with the requirements 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. 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 Rancho Seco License. The new license condition would incorporate the NRC approved License Termination Plan (LTP) into the Rancho Seco license, and allow the licensee to make certain changes to this approved LTP without prior NRC review or approval. The new License Condition would appear as follows:

(4) License Termination Plan (LTP)

NRC License Amendment No. 133 approves the License Termination Plan.

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 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

[1] ADAMS Accession Number ML051020434

[2] ADAMS Accession Number ML061460053

[3] ADAMS Accession Number ML062220351

[4] ADAMS Accession Number ML071580233

(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.

Re-classification 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, re-classification 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 October 24, 2006[5]. The licensee responded by letter dated November 21, 2006[6]. The licensee also provided supporting information in "Hydrogeological Characterization of Rancho Seco"[7] (Hydrogeological Characterization Report), dated March 15, 2006, and "Rancho Seco Groundwater Monitoring Report"[8] (Groundwater Monitoring Report), dated September 6, 2006.

2.0 EVALUATION

The licensee submitted its LTP in accordance with 10 CFR 50.82(a)(9), that requires the LTP to contain the following information: (1) a site characterization; (2) identification of remaining dismantlement activities; (3) plans for 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 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 (Ref. 1), 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.

The LTP describes SMUD's approach for demonstrating compliance with radiological criteria, for unrestricted use. As stated in 10 CFR 20.1402, the annual dose limit is 0.25 mSv (25 mrem) total effective dose equivalent (TEDE) above background from all pathways, including ground water; SMUD must also reduce residual radioactivity to "as low as is reasonably achievable" (ALARA) levels.

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[5] ADAMS Accession Number ML062850386

[6] ADAMS Accession Number ML063330062

[7] ADAMS Accession Number ML060810160

[8] ADAMS Accession Number ML062980496

2.1 Site Characterization

2.1.1 Facility Radiological Status

Site characterization surveys are conducted 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 remediation activities; confirm the effectiveness of 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.

Ranch Seco conducted site characterization activities that included a historical site assessment (HSA), pre-characterization scoping survey, and a characterization survey. The requirements for a site characterization are defined in 10 CFR 50.82(a)(9)(ii)(A).

Rancho Seco conducted an HSA that included a review of plant incident records, plant maintenance records, plant modification records, plant radiological survey records, and regulatory reports submitted by SMUD to various governmental agencies, as well as written questions and oral interviews with current and past employees to identify all locations, both inside and outside the facility, where radiological spills, disposals, operational activities, or other radiological accidents could have resulted in contamination. In Table 2-1 of the LTP, interviewees identified 146 observations where there may have been a radiological impact that was not already identified on a list provided to the interviewee. Historical site aerial photographs and physical inspections of the plant were conducted to verify and validate the historical assessment and a site reconnaissance was conducted to verify locations and current conditions of items or issues discovered during these investigations.

Summaries of the original shutdown and current radiological/non-radiological conditions of the site are provided in the LTP as Table 2-2 that shows the history of the operational activities. Section 2.1.8 provides a summary of the area radiological impact. Figure 2-2 identifies the impacted and non-impacted areas. Figure 2-3 provides the area designation and Table 2-5 identifies which areas in Figure 2-3 are impacted and non-impacted. It is noted that two survey units within Area 5 are defined as impacted, although Area 5, in general, is defined as a non-impacted area.

As of January 2006, the decommissioning project has removed many major systems and components (with the exception of embedded and buried piping) in the Turbine Building, Auxiliary Building, Fuel Handling Building, Containment Building, as well as the removal of temporary buildings and structures outside of the power block. A majority of these systems and components will not be included in the FSS.

Several areas within the Industrial Area have been identified as having been radiologically impacted by the operation of the facility including the retention basins, tank farm, barrel farm, areas adjacent to the Regenerative Hold Up Tank Area (RHUT), storm drains, oily water separator, cooling tower basins, and the turbine building drains and sumps. There are several areas outside of the Industrial Area, identified as the Non-Industrial Area, that have historically had radionuclide concentrations detected above background. These areas include the discharge canal sediment, discharge canal soil, depression area soil, and the storm drain outfall.

SMUD identifies a list of 26 radionuclides in Table 6-1 of the LTP that are considered to be site specific and are potentially present on the remaining site structural surfaces, in site soils, and groundwater following completion of the decommissioning activities. A total of 23 radionuclides were identified in the LTP as contributing less than 0.1 percent of total activity. These radionculdes are Cl-36, Ar-39, Ca-41, Mn-53, Se-79, Kr-81, Kr-85, Zr-93, Mo-93, Sn-121m, I-129, Ba-133, Cs-135, Pm-145, Sm-146, Sm-151, Tb-158, Ho-166m, Hf-178m, Pb-205, U-233, Am-243, and Cm-243. In 2004, the licensee identified a spent fuel pool cooler pad soil sample as having the highest level of contamination of any soil sample collected during the characterization process. This soil sample was submitted to an offsite vendor laboratory for analysis. The offsite vendor laboratory was asked to examine the sample for all 26 radionuclides. The results detected only 6 radionuclides above the minimum level of detection. These radionuclides were C-14, Co-60, Ni-63, Sr-90, Cs-134, and Cs-137. These radionuclides were used to derive the single nuclide derived concentration guideline levels (DCGLs), and will form the basis for the FSS. Additionally, 75 volumetric samples representing contaminated structure surfaces were examined and 8 were selected for further analysis by an offsite vendor laboratory. The offsite vendor laboratory reported positive results for the following hard-to-detect (HTD) radionuclides, H-3, C-14, Fe-55, Ni-59, Ni-63, Sr-90, Tc-99, Pu-238, Pu-239, Pu-240, Pu-241, and Am-241 as well as the following easy to detect radionuclides, Co-60 and Cs-137. The easy to detect radionuclides (Co-60 and Cs-137) represented 90% or more of the individual sample nuclide fraction. The results were used to determine the gross beta DCGL for structures.

The extent and range of radiological contamination of structures and surfaces are listed in the LTP in Table 2-17, Table 5-4A (General Open Land Areas), Table 5-4B (Site Surface Soils), Table 5-4C (Paved Surfaces and Foundation Pads), Table 5-4D (Structures), and Table 5-4E (Remaining Buried and Embedded Piping). Table 2-8 identifies the groundwater monitoring radiochemical results.

The instrumentation that was used for the characterization study will also be used for the FSS. Table 2-13 of the LTP list the instruments and detectors, the type of radiation detected, the sample and background count time, background count rate, instrument efficiency, and the minimal detectable concentration for static and scan counts.

The instruments were calibrated using National Institute of Standards and Technology traceable sources, approved procedures and by trained and qualified personnel. Instruments were source checked prior to, and after, each survey. A fraction of the volumetric samples were collected as duplicates as part of the quality control program. The date quality objectives (DQO) format was used to develop the characterization survey. The DQOs for the site characterization included identifying the types and quantities of media to collect. A summary of the types and quantities of media collected from surfaces/structures, environs, and buried/embedded piping are outlined below:

Surfaces and Structures,

- Approximately 2000 beta scans and direct measurements were acquired from structure pads and asphalt surfaces including roadways.

- 87 concrete scabble samples were collected from the Auxiliary Building
- 26 scabble samples were collected from the Spent Fuel Pool floor and walls
- 11 concrete core samples were collected through the entire thickness of the Spent Fuel Pool walls
- 19 core samples were collected from the Containment Building
- 5 core samples were collected from different elevations of the Containment Building
- 5 concrete scabble samples were collected from the Turbine Building including the condenser pit floor and walls.

Environs

- Over 3000 gamma scans and direct measurements taken in impacted and non-impacted areas
- Collection of over 600 soil and sediment samples

Buried and Embedded Piping

- 18 samples from the interior of the Turbine Building embedded drain piping and sumps
- 5 samples of RHUT piping
- 19 samples from the interior of the Auxiliary Building drain pipings
- 13 samples from embedded piping associated with the Spent Fuel Pool
- 4 samples were collected from the Containment Building drains and systems.

A number of site studies have been conducted at Rancho Seco over the years that provided a sufficient source of data for background. Because of the relatively large DCGLs, Rancho Seco expects not to apply background subtraction or background references to the FSS. With the exception of Cs-137, many of the radionuclides identified in the characterization study do not occur naturally in background. According to the Multi-Agency Radiation Survey And Site Investigation Manual (MARSSIM), if the radionuclide contaminants of interest do not occur in background, or the background levels are known to be a small fraction of the DCGLw (i.e., <10%), the survey unit radiological conditions may be compared directly to the specified DCGL and reference area background surveys are not necessary.

Rancho Seco commits to continuing characterization in the LTP. The results of future characterization will be compared and evaluated against the current list of radionuclides identified for characterization and its radionuclide fractions, and revised, if necessary, to classify structures, soils, and other site media.

The staff finds the site characterization program acceptable, based on the information described in the LTP. 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.

2.1.2 Geology

The Rancho Seco site is located within the Great Valley Geomorphic Province, which is a wide structural trough bounded by the Sierra Nevada mountains on the east and the Coast Range

mountains on the west. The stratigraphy of the site from youngest to oldest deposits consists of the following deposits:

- Recent alluvium consisting of stream-deposited gravel, sand, and silt. This material is located within present drainage and ranges in thickness from 0 to 5 feet.

- Older alluvium consists of older stream and terrace deposits of gravel, sand, and silt with a thickness of 0 to 20 feet. This material covers the floodplains and terraces in the southwestern portion of the site. It contains deposits of well-rounded cobbles, pebbles, and sand derived from pre-Cretaceous sediments on pediment surfaces.

- The Laguna Formation consists of sand, silt, and some gravel, and it may contain clay. It contains poorly bedded materials of silicic volcanic origin. This formation occurs at the surface across much of the site with its bottom depth of 130 feet below ground surface (bgs).

- The Mehrten Formation consists of fluviatile sandstone, siltstone, and conglomerate composed primarily of andesitic detritus with local horizons of coarse andesitic agglomerate of mudflow origin. This formation is found at the surface west of the industrial area has an approximate thickness of 225 feet bgs.

- The Valley Springs Formation of volcanic origin consists of pumice, siliceous ash, clay, vitreous tuff, glassy quartz sand, and conglomerate. It is well bedded and derived from rhyolithic volcanic eruptions from the Sierra Nevada mountains. This formation has no surface exposure at the site, and it has an estimated thickness of 250 feet.

- The Ione Formation contains clay, sand, sandstone, and conglomerate. It has an estimated thickness of 200 to 400 feet beneath the site, and it is not exposed at the land surface. No onsite borings penetrate either the Valley Springs or the Ione Formations. It is estimated that below the Ione Formation are metamorphic basement rocks at about 2,000 feet bgs.

No faults have been observed within 10 miles of the site, and the only observed structure in the sedimentary rocks is a gentle western dip of 1 to 3 degrees caused by gradual uplift of the Sierra Nevada.

2.1.3 Groundwater

The NRC staff has evaluated the following: (1) the extent that groundwater at the Rancho Seco site contains plant-generated radionuclides, (2) whether groundwater contaminated with plant-generated radionuclides has migrated offsite, and (3) the impact that this potentially contaminated groundwater has upon potential receptors. This evaluation is based upon Rancho Seco's LTP and supporting documents and on NRC's independent assessment.

Groundwater Regime. The Rancho Seco site is located within the Cosumnes Subbasin of the San Joaquin Valley Groundwater Basin. This subbbasin has both unconsolidated and semi-consolidated sedimentary deposits that may have water-bearing units along with the fractures in

the underlying basement rocks. However, at this site the uppermost water-bearing units are within the Mehrten Formation. The shallowest water-bearing unit is the water table at approximately 165 feet bgs. It is possible that the water-bearing units may exist through the Ione Formation to a depth of 2,000 feet bgs. However, this has not been verified because no borings or wells have been extended below the Mehrten Formation. The general groundwater flow direction in the Cosumnes Subbasin is westward. However, the groundwater flow direction for the water-bearing units within the Mehrten Formation at the site is southwest.

The Mehrten Formation's groundwater yield is lower beneath the site than at other locations in the Cosumnes Subbasin. The predominant lithologies of the water-bearing units within the Mehrten Formation are siltstones and claystones. The hydraulic conductivity (K) of these lithologies range from $1x10^{-7}$ to $1x10^{-4}$ cm/sec.

There is a groundwater depression in this water-bearing unit within the Mehrten Formation southwest of the site near Galt, California because municipal and irrigation wells in this area are pumped at rates exceeding the recharge capacity of this unit. If groundwater development near Galt continues, the groundwater depression may extend to the Rancho Seco site. Therefore, it is important to determine whether plant-generated radionuclides have reached the water-bearing unit beneath the Rancho Seco site because receptors (humans drinking water) near Galt and between Galt and the site may eventually extract this groundwater.

Radiological Spills, Leaks, and Releases. SMUD in its LTP and its Historical Site Assessment has acknowledged that spills, leaks, and releases of plant-generated radionuclides have occurred at the Rancho Seco site. During the operation of the nuclear reactor, leaks, spills, and/or releases occurred that impacted the Spent Fuel Building, Spent Fuel Cooler Pad outside the Spent Fuel Building, Tank Farm, Retention Basins, Barrel Farm, storm drains, Turbine Building drains and sumps, Oily Water Separator, and Regenerant Hold Up Tank areas. The potential for radionuclide movement to the saturated groundwater was significantly greater for leaks associated with the spent fuel building and spent fuel cooler pad than with the other mentioned structures and areas. Remediation of soils in the Spent Fuel Building and Spent Fuel Cooler Pad has been completed. SMUD has not observed plant-generated radionuclides at the Spent Fuel Building at depths greater than 25 feet below grade.

Radiological Monitoring Wells. In 2005, SMUD addressed deficiencies in its radiological monitoring program by installing four nests of monitoring wells within and down gradient from the Industrial area. Prior to the installation of these wells, there were no monitoring wells in the Industrial area that were designed for monitoring plant-generated radionuclides. However, there are two water supply wells, SW-1 and SW-2, that are located up gradient of the Industrial area and within the Industrial area, respectively, and four observation wells; OW-1, OW-2, OW-3, and OW-3a; that are located down gradient from the Industrial area where the evaporation pond investigation was performed. All of these wells are delineated in Figure 2-1 of the Hydrogeological Characterization Report.

In May and June of 2005, the four nests of three monitoring wells: MW1A-C, MW2A-C, MW3A-C, and MW4A-C, were installed within and down gradient of the Industrial area. These wells were all screened in water-bearing units of the Mehrten Formation from about 160 to 340 feet bgs with the lowered screened well at least 30 feet below the bottom screen of the well immediately above it. SMUD performed four quarterly sampling events: summer and fall of 2005

and winter and spring of 2006, where groundwater samples were analyzed for potential plant-generated radionuclides.

Potentiometric Groundwater Surfaces and Groundwater Flow Directions. The groundwater levels in the four nests of monitoring wells indicate that there are multiple water-bearing units between the water table (about 165 feet bgs near the Industrial area) and 300 feet bgs in this area, that the groundwater flow direction is toward the southwest, and that the vertical hydraulic gradient between the wells in each nest is usually downward but may vary over time. A potentiometric surface map developed from monitoring wells MW1B, MW2B, MW3B, and MW4B and observation wells OW-2 and OW-3, which are screened at similar intervals levels to the B level monitoring wells. The groundwater levels measured on December 6, 2005 are delineated in Figure 2-7 in the Hydrogeological Characterization Report as the potentiometric surface for B level monitoring wells. An examination of Table 2-1 in the Hydrogeological Characterization Report indicates that the potentiometric surfaces for groundwater levels measured in March and June 2006 had similar general trends (that is, the groundwater flow direction was toward the southwest). The potentiometric surfaces for the A and C level monitoring wells for these same time periods also demonstrated groundwater flow directions to the southwest.

Aquifer Parameters. As mentioned above, the predominant lithologies of the water-bearing unit within the Mehrten Formation are siltstones and claystones, which have hydraulic conductivity (K) values that range from $1x10^{-7}$ to $1x10^{-4}$ cm/sec. However, sand zones within the water bearing unit have K values that range between $1x10^{-5}$ to $1x10^{-3}$ cm/sec. These estimates are based upon packer tests performed on selected intervals of boreholes and observation wells in the geotechnical investigation of the proposed evaporation ponds about 0.25 miles southwest of the Industrial area in 1985. Thus, the K values for the screened intervals of the four nests of wells installed in 2005 (one additional well was installed in 2006) would fall within the $1x10^{-5}$ to $1x10^{-3}$ cm/sec range.

Infiltration and Recharge. The average annual precipitation is approximately 16.3 inches at Sacramento, California with most of precipitation occurring during October through May, the rainy season. Recharge to the saturated zone will occur from infiltration of precipitation, and the NRC staff believes that this recharge will likely average less than 0.5 inch per year because of the low precipitation and high evapotranspiration rates in this area. Thus, the major components of recharge in this area are from surface water along active channels and from deep percolation of applied irrigation water.

*Groundwater Sampling and Analysis for Radionuclides***.** SMUD performed four groundwater sampling events in the third and fourth quarters of 2005 and the first and second quarters of 2006 to evaluate potential plant-generated radionuclides in the groundwater beneath the Industrial area and nearby areas. The shallowest monitoring well in each nest [MW1C (only for the third and fourth quarters of 2005), MW1D (only for the first and second quarters of 2006), MW2A, MW3A, and MW4A] that yielded groundwater was sampled for each sampling event. SMUD had General Engineering Laboratories (GEL) perform the following radionuclide analyses on unfiltered samples for the third and fourth quarters of 2005 (and for the first quarter of 2006 for MW1D): gross alpha, gross beta, H-3, C-14, Na-22, Fe-55, Ni-59, Co-60, Ni-63, Sr-90, Nb-94, Tc-99, Ag-108m, Sb-125, Cs-134, Cs-137, Pm-147, Eu-152, Eu-154, Eu-155, Np-237, Pu-238, Pu-239/240, Pu-241, Am-241, Pu-242, and Cm-244. For the first and second quarters of 2006, SMUD personnel performed onsite analyses by liquid scintillation for H-3 and gamma

spectrometry isotopic analyses for Co-60, Ag-108m, Cs-134, and Cs-137 on samples from the above monitoring wells (URS Corporation, 2006a). The analytical results for theses analyses indicated that no U.S. EPA Primary Drinking Water Standards were exceeded and that all the results except for five gross beta and two gross alpha results were less than the *a posteriori* calculated minimum detection activity.

Although no background monitoring wells were sampled for the above potential plant-generated radionuclides during the four sampling events, SMUD did analyze the above samples and groundwater samples from water supply wells SW1, SW2, and RSPW for naturally occurring radionuclides. SMUD personnel performed onsite gamma spectrometry analysis on the above mentioned samples for the following naturally occurring radionuclides: K-40, Pb-214, Bi-214, Rn-222, Ra-226, Ra-224, and U-235 for all four quarters. The analytical results for these analyses indicate that wells hydraulically up gradient from the Industrial area, (that is, water supply wells SW1 and RSPW) have similar radiochemistry to the monitoring wells.

Synoptic ground-water level measurements were also performed on all monitoring wells for each sampling event. Table 5 lists the groundwater elevations for the wells.

The NRC did not collect split groundwater samples from any of the monitoring wells at this site. The NRC staff concluded that this was acceptable based upon the above radiological results (no plant-generated radionuclides were observed in the groundwater), SMUD's sampling and analytical procedures, and GEL's analytical procedures.

Rate of Groundwater Movement. The movement of groundwater and potential plant-generated radionuclides dissolved in the groundwater is extremely slow. The movement of groundwater downward from the site to the saturated zone, the uppermost water-bearing unit at about 165 feet bgs, is estimated by SMUD to take 80 years based upon a vertical hydraulic conductivity of 2.0×10^{-4} cm/sec. SMUD also estimated the groundwater travel time for groundwater beneath the Industrial area to reach the site boundary, a distance of 3,100 feet at 1,500 years using a horizontal hydraulic conductivity of 2.0×10^{-4} cm/sec. The NRC staff has concluded, based on the estimated K ranges for the unsaturated zone and the water-bearing units, that these travel time estimates for groundwater are reasonable. Also, the movement of most potential plant-generated radionuclides dissolved in the groundwater would be slower.

Conclusions. Currently, no plant-generated radionuclides have been observed in the groundwater beneath the Rancho Seco site. Also, no plant-generated radionuclides have been observed in the soil or rock materials beneath the Industrial area at a depth greater than 25 feet. Thus, the likelihood in the future that plant-generated radionuclides will reach the groundwater or migrate offsite is extremely small. Therefore, there is no current or expected future concern that plant-generated radionuclides dissolved in the groundwater will impact receptors.

The NRC staff has concluded that the SMUD's groundwater characterization with respect to potential plant-generated radionuclides is acceptable.

2.1.4 Surface Water

Surface water in the vicinity of the site includes Clay Creek; unnamed tributaries to Clay Creek; Rancho Seco Reservoir, which was formed by damming Clay Creek in the southeast area of the

SMUD when Rancho Seco reactor was constructed, and an area of vernal pools and seasonal marches. All of these features are south or southeast of the Industrial area. Clay Creek eventually discharges into Hadselville Creek offsite. Hadselville Creek is a tributary of Laguna Creek South, which flows into the Cosumnes River.

Runoff from the Industrial area drains into an unnamed tributary of Clay Creek. Also, releases from the Industrial area, averaging 6,000 gallons per minute, are discharged in the liquid effluent pathway downstream from the site's retention basins into this creek. Most of these releases (volume of surface water) to this creek are conveyed to the site from the Folsom South Canal. Other sources of flow in this unnamed creek are releases from the Rancho Seco Reservoir and runoff from its catchment west of the dam and up gradient from the Industrial area.

Since the investigation for the development of Rancho Seco in the 1960's, no flooding has occurred within the site boundaries from storm runoff. Also, the Industrial area is not within the 100-year floodplain. However, vernal pools and seasonal marshes develop west of the Industrial area in shallow surface depressions during and after the December to March rainy season.

There is no indication that plant-generated radionuclides have impacted the surface waters of Clay Creek or its tributaries above permitted effluent criteria.

The NRC staff has determined that the SMUD's surface water characterization is acceptable.

2.1.2 Site Characterization - Summary Finding

The staff has reviewed the information in the LTP for Rancho Seco, according to Section B.2 of NUREG-1700 (Ref. 30). 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).

2.2 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 December 31, 2005. The information included those areas and equipment that need further remediation and an estimate of radiological conditions that may be encountered. Included are estimates of associated occupational radiation dose and projected volumes of radioactive waste.

The licensee provided an overview and describes the major remaining components of contaminated plant systems and, as appropriate, a description of specific equipment remediation considerations. Information related to the remaining decommissioning and dismantlement (D&D) tasks is also provided. This information included an estimate of the quantity of radioactive material to be released in accordance with 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 final state of the Industrial Area will be a partially abandoned facility with portions, other than the power block, available for reuse. The

impacts of decommissioning activities performed will be to reduce residual radioactivity to a level that permits release of the property for beneficial reuse by SMUD for industrial purposes.

Major activities remaining include reactor vessel internals removal, rector vessel removal, removing certain underground piping, reactor building internal structures removal, embedded piping decontamination, auxiliary building decontamination, spent fuel pool decontamination, wastewater systems decontamination, and reactor building decontamination.

The total nuclear worker exposure during decommissioning is currently estimated to be less than 200 person-rem. This estimate is significantly below the 1,215 person-rem estimate of the Generic Environmental Impact Statement (GEIS) (NUREG-0586) for immediate dismantlement and below the ten-year SAFSTOR estimate of 664 person-rem.

Rancho Seco has shipped for radioactive disposal approximately 5,560 cubic meters (196,325 cubic feet) through December 31, 2005. The estimate of remaining waste is 11,730 cubic meters (414,206 cubic feet), most of which is very low activity concrete debris from the Reactor Building interior. This volume of waste is bounded by the GEIS typical volume of 18,340 cubic meters (647,700 cubic feet) for the reference pressurized water reactor.

Currently SMUD intends to defer disposal of Class B and C waste. As a result, that waste will be stored in the Interim Onsite Storage Building (IOSB) until such time as an acceptable waste disposal site is available. Once a solution is available, waste will be shipped and the building will be decontaminated as required. Once the IOSB is decontaminated a FSS will be performed in accordance with the LTP and a final release from the Part 50 license will be requested. The time frame for that request is currently scheduled for 2028.

Currently, no permanent buildings or structures on site are scheduled for demolition. The switchyard is in use for the Cosumnes Power Plant, the IOSB will be used for storage of Class B and C waste, the Administration Building is used as an Emergency Operation Facility for SMUD in case Sacramento facilities are unavailable and the Secondary Alarm Station is used by ISFSI security personnel. Various other buildings may be used for office space or maintenance activities. The District may, at some future date, decide to demolish or refurbish any of the buildings or structures onsite. However, the impacted structures (Reactor Building, Spent Fuel Building and the Auxiliary Building) are unlikely to be reused or demolished in the near future. Many possible uses for the site, or portions of the site, have been considered but it will remain SMUD property and the site's most likely use would be for future electric generation due to switchyard access and water availability.

The staff has reviewed the LTP against the information in Section B.3 of NUREG-1700. 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).

2.3 Plans for 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 an industrial use scenario.

Remediation techniques that may be used for the structural surfaces include washing, wiping, pressure washing, vacuuming, scabbling, chipping, and sponge or abrasive blasting. Washing, wiping, abrasive blasting, vacuuming and pressure washing techniques may be used for both metal and concrete surfaces. Scabbling and chipping are mechanical surface removal methods that are intended 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.

Soil contamination above the site specific DCGL will be removed and disposed of as radioactive waste. Soil remediation equipment will include, but not be limited to, back and track hoe excavators. As practical, when the remediation depth approaches the soil interface region between unacceptable and acceptable contamination, a squared edge excavator bucket design or similar technique may be used. This simple methodology minimizes the mixing of contaminated soils with acceptable lower soil layers as would occur with a toothed excavator bucket. Remediation of soils will include the use of established Excavation Safety and Environmental Control procedures. Additionally, soil handling procedures and work package instructions will augment the above guidance and procedural requirements to ensure adequate erosion, sediment, and air emission controls during soil remediation.

The Radiation Protection Program approved for decommissioning is similar to the program in place during commercial power operation. During power operations, contaminated structures, systems and components were decontaminated in order to perform maintenance or repair actions. The techniques used during operations are the same or similar to the techniques used during decommissioning to reduce personnel exposure to radiation and contamination and to prevent the spread of contamination from established contaminated areas.

As specified in Part 20, Subpart E, a site is acceptable for unrestricted use if the remaining residual radioactivity results in a TEDE less than or equal to 0.25 mSv (25 mrem) per year above background, and the remaining residual radioactivity is reduced to levels that are ALARA. The licensee provided its ALARA analysis process in Section 4.4 of the Rancho Seco LTP. 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 Section B.4 of NUREG-1700 and against similar decommissioning activities conducted at other plants. 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 the remediation plans for the remaining decommissioning activities.

2.4 Final Radiation Survey Plan

The Rancho Seco LTP describes the FSS plan for demonstrating that the plant and site will meet the proposed release limits. The regulations applicable to this area of review are 10 CFR 50.82(a)(9)(ii)(D) and 10 CFR 20.1501(a) and (b). The purpose of the final status survey is to demonstrate that each area, as defined by survey classifications, meets the radiological crtieria for license termination.

This section reviews the identification of major radiological contaminants, methods used for addressing hard-to-detect radionuclides, access control procedures to control recontamination of clean areas, the Quality Assurance Program and the Quality Assurance Project Plan and methods for surveying embedded piping.

The identification of major radiological contaminants are listed in Table 6-1 in the LTP. The list contains 26 radionuclides that are potentially present at Rancho Seco. Radionuclides with half lives of less than 2 years were not included in the list. Radionuclides with half lives of two years or less would have decayed substantially since plant shutdown in 1989.

Surrogate ratio DCGLs are used to express a mixture of radionuclides which may include hard-to-detect radionuclides, as a single value. Equation 5-1, on page 5-8, in the LTP is the method available for surrogate ratios. This method will be applied to soils and materials. The licensee will collect data prior to remediation and will review post remediation and FSS data to ensure that the surrogate ratios are still appropriate. The surrogate ratio will be evaluated using the Rancho Seco DQO and will be modified, if necessary.

The Rancho Seco access control procedure consist of a turnover, walkdown, transfer of control and the isolation and control of the survey unit(s). For those areas where the potential for recontamination may occur, Rancho Seco will implement the following control measures: 1) personnel training, 2) installation of barriers to control access to surveyed areas, 3) installation of barriers to prevent the migration of contamination from adjacent or overhead areas from water runoff, etc. 4) installation of postings requiring contamination monitoring prior to surveyed area access, 5) locking entrances to surveyed areas of the facility, 6) installation of tamper-evident devices at entrance points, or 7) routine surveys to monitor and verify adequate isolation and control measures. The licensee will not perform routine surveys of open land areas that are not normally occupied and are unlikely to be impacted by decommissioning activities. Post FSS survey locations will be judgementally selected for survey, based on technical or site specific knowledge and current conditions present in or near the survey area.

The Rancho Seco FSS Quality Assurance Project Plan was developed to ensure that the DQO process for the FSS design, analysis, and evaluation are met. The plan ensures that the elements of the final status survey plan are implemented in accordance with the approved procedures, 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 Quality Control (QC) and management oversight provided, and corrective actions, if any, are implemented in a timely manner.

The Quality Assurance Plan and the Quality Assurance Project Plan identify critical positions of the organization, including the functional areas of responsibility and descriptions. The Dismantlement Superintendent has overall responsibility for the program direction, technical content, and ensuring compliance. The Technical Specialist for FSS may include technical support and development of FSS procedures, design of final status surveys, preparation of

survey execution instructions, development of specific technical analysis documents supporting FSS activities, and the review of survey packages and data collected in the field. The Work Plan Coordinator develops work instructions using the work order process.

The FSS Field Coordinators are responsible for control and implementation of survey packages during field activities. These duties include coordination of turnover surveys, survey area preparation (i.e., gridding), ensuring FSS sampling is conducted in accordance with applicable procedures and work instructions, maintaining access control over completed FSS survey areas, determining survey area accessibility requirements, coordinating and scheduling of FSS technicians and ensuring all necessary instrumentation and equipment are available.

The FSS activities are conducted in accordance with written procedures and instructions. These procedures and instructions are identified by title in Section 5.9.1 of the Rancho Seco LTP. RSAP-1702, Quality Assurance for Radiological Monitoring Program, addresses water, air, and soil sampling procedure(s). Training and qualification includes procedures governing the conduct of the FSS program, operation of field and laboratory instrumentation, and the collection of FSS measurements and samples. Each FSS measurement will be identified by individual, date, instrument, location, type of measurement, and mode of operation.

Rancho Seco plans to institute quality control checks which include a minimum of 5% of randomly selected Class 1, 2, or 3 survey units performed by a different technician with results compared to the original survey results. This general process will be used for structures and systems as well as soils, water, and sediments. If QC replicate measurement or sample results fall outside of their acceptance criteria, an investigation will be performed and documented and corrective action will be implemented, if necessary. Other elements of the Rancho Seco Quality Assurance Program and Quality Assurance Project Plan include instrumentation selection, calibration and operation, chain of custody, control of consumables, control of vendor supplied services, database control, and data management.

Rancho Seco will use a dual method of assessments for the FSS program. This will include self-assessments by the FSS staff as well as independent QC assessments in accordance with the Quality Assurance Program. Quality Assurance personnel plan to randomly select approximately 5% of the survey packages from survey units for independent review. Survey data will be reviewed prior to evaluation or analysis for completeness and for the presence of outliers. Comparison to investigation levels will be made by the Rancho Seco staff and measurements exceeding the investigation levels will be evaluated. Reports of audits and trend data will be reported to management.

Methods used by Rancho Seco to survey embedded piping depend on whether the internal surfaces have been remediated or have not been remediated. Where no remediation has been conducted, inaccessible or difficult areas are assumed to have the same level of residual radioactivity as that found on accessible internal surfaces. Where remediation has taken place, representative samples of the inaccessible internal surfaces are obtained, an assessment of pre-remediation survey data is performed, so that a reasonable approximation of the residual radioactivity on the inaccessible internal surfaces can be made. Instrumentation used for pipe surveys are identified in Table 5-11 and Table 5-12 of the LTP. Sensitivity levels for pipe survey instrumentation are indentified in Section 5.4.3.4.7 of the LTP.

The probability of making decision errors is evaluated by statistical methods. The licensee will use survey results to evaluate the condition of the environment against the *null hypothesis*. The *null hypothesis* is: "The residual radioactivity in the survey unit exceeds the release criterion." A Type I decision error occurs when the *null hypothesis* is rejected when it is true. Therefore a Type I decision error would determine that an area was acceptable for release when it is not. Appendix E of NUREG-1757, Volume 2 recommends using a Type I error probability of 0.05. Following the NUREG-1757, Volume 2 guidance, the licensee will set the Type I error probability at 0.05.

The staff finds the FSS plan program acceptable, based on the information described in the LTP. 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.

2.5 Compliance with Radiological Criteria for License Termination

Chapter 6 of the LTP discusses the development of residual radionuclide concentration levels that will be used to demonstrate compliance with the regulations for releasing the site for unrestricted use. The primary scenario considered in developing radionuclide-specific DCGLs for the Rancho Seco facility is the industrial worker scenario. Rancho Seco also considered two alternate exposure scenarios and evaluated the potential dose impacts for comparison to the primary scenario: 1) a resident farmer scenario and 2) a cattle grazing scenario.

2.5.1 Site Release Criteria

The licensee proposes unrestricted release of the site in compliance with the requirements of 10 CFR 20.1402. Thus, the residual radioactivity that is distinguishable from background must not cause the TEDE to an average member of the critical group to exceed 25 mrem/year (0.25 mSv/year). Residual radioactivity must also be reduced to levels that are ALARA.

As required under 10 CFR 20.1402, expected doses are to be evaluated for the average member of the critical group, which is not necessarily the same as the maximally exposed individual. The use of the "average member of the critical group" acknowledges that any hypothetical "individual" used in the dose assessment is based, in some manner, on the statistical results from data gathered from groups of individuals. DCGLs have been developed as acceptable levels of residual radioactivity that can be left at the site in compliance with the unrestricted release criteria. Development of DCGLs that will be used to demonstrate compliance with the regulations are provided in Section 6.6 of the LTP.

Calculating the dose to the critical group is intended to bound the individual dose to other possible exposure groups because the critical group is a relatively small group of individuals, due to their habits, actions, and characteristics, who could receive among the highest potential dose at some time in the future. By using the hypothetical critical group as the dose receptor, it is unlikely that any individual would actually receive doses in excess of that calculated for the average member of the critical group.

2.5.2 Source Term

The licensee developed a site-specific suite of 26 potential radionuclides listed in Table 2.5.1. The listed suite of radionuclides is considered to be potentially present in site soil, structural surfaces, and in groundwater following decommissioning activities. The DCGLs were derived for those radionuclides that contribute to site contamination, based on a screening analysis which considered half-life, detection of radionuclides at the site, and the potential contribution from each radionuclide to the overall dose. A comprehensive suite of approximately 52 radionuclides were first identified as potential radionuclides by using NUREG/CR-3474, "Long-Lived Activation Products in Reactor Materials" and NUREG/CR-4289, "Residual Radionuclide Contamination Within and Around Commercial Nuclear Power Plants." Thereafter, additional radionuclides were identified in NUREG/CR-0130, "Technology, Safety and Cost of Decommissioning." Only those radionuclides with half-lives of two or more years were included in the suite. Radionuclides with half-lives less than two years would not be observed since seven or more half-lives have occurred since the final shutdown of the reactor. As a final step in the source term development process, the licensee used the ORIGIN computer code to determine if there were additional radionuclides that should be added to the suite.

The licensee discounted radionuclides from the suite which contributed less than 0.1 percent of the total activity provided that the potential dose contributed by the sum of the radionuclides discounted did not exceed one percent of the total calculated dose. The inventory for each radionuclide was determined from activity inventories published in NUREG/CR-3474. Based on this criterion of contributing less than 0.1 percent of the total activity, 23 radionuclides were discounted from the suite. The list of discounted radionuclides is provided in Section 6.3.2 of the LTP.

To evaluate compliance with the dose criteria for discounted radionuclides, the licensee used the DandD Version 2.1.0 computer code to calculate doses for both residential and building occupancy scenarios. For those radionuclides not supported by the DandD computer code, the licensee evaluated the potential dose contribution by comparison of the inhalation and ingestion exposure-to-dose conversion factors in Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion."

Radionuclides were also discounted based on waste stream characterization analyses. Radionuclides listed in 10 CFR 61.55 Tables 1 and 2 as well as other supplementary radionuclides on a select basis were discounted if they were not identified by the waste stream analysis. An additional radionuclide, Pu-242, was identified by the waste stream analysis and was added to the suite of radionuclides. The naturally occurring radionuclides (K-40, U-234, U-235, U-236 and U-238) which were not detected in characterization survey samples at concentrations distinguishable from naturally occurring concentrations were discounted from the site-specific suite. Staff reviewed the approach used by the licensee for screening out radionuclides and concluded that the resulting suite of radionuclides for which DCGL values are established is reasonable and appropriate for this site.

Table 2.5.1 Site-Specific Suite of Radionuclides developed for Rancho Seco site

H-3	Sr-90	Pm-147	Pu-240
C-14	Nb-94	Eu-152	Pu-241

Na-22	Tc-99	Eu-154	Pu-242
Fe-55	Ag-108	Eu-155	Am-241
Ni-59	Sb-125	Np-237	Cm-244
Co-60	Cs-134	Pu-238	
Ni-63	Cs-137	Pu-239	

2.5.3 Exposure Scenarios

The licensee considered an industrial worker scenario for surface and subsurface soil. An industrial worker scenario is also considered for potential exposure to contaminated building structures, bulk materials, and embedded piping. Scenario definition and exposure pathways are described in the following two sections.

2.5.3.1 Surface and Subsurface Soil Scenario and Exposure Pathways

For surface and subsurface soil, the licensee considered an industrial worker to represent the average member of the critical group. The worker is assumed to be exposed to contaminated soil by the following exposure pathways: 1) direct exposure, 2) inhalation of airborne radionuclides, 3) ingestion of contaminated soil, 4) drinking water from a contaminated well, and 5) exposure to buried piping.

The justification for the industrial worker scenario is described in detail in section 6.4.2.1 of the LTP. Scenario justification included the following: 1) the entire site is owned by the District and is not a tax burden, 2) the District has no plans to release all or part of the 2,480 acre (1.0×10^7 m^2) site for ownership by members of the public, 3) the Rancho Seco switchyard is the District's major electrical distribution center with the Western Grid transmission system, 4) a 50-acre (2.0×10^5 m^2) photovoltaic generating facility is located on a non-impacted portion of the site, and 5) a new 500 MWe natural gas fueled combined-cycle generating facility is located on a 30-acre (1.2×10^5 m^2) portion of the 2,480-acre (1.0×10^7 m^2) site and began commercial operation in February 2006. Staff reviewed the justification for selecting the industrial worker scenario and found the scenario justification reasonable and acceptable. The scenario, critical group determination, and exposure pathways are consistent with the guidance provided in NUREG-1757, Volume 2.

2.5.3.2 Building Occupancy Scenario and Exposure Pathways

For building surfaces, the industrial worker was considered to represent the average member of the critical group. The building occupancy scenario is used to evaluate potential exposure to fixed and removable surface radioactivity within structures that will be left on the site after license termination. The worker is assumed to be exposed to penetrating radiation from surface sources, inhalation of resuspended surface contamination, and inadvertent ingestion of surface

contamination. As stated in the LTP, concrete structures will remain in place after equipment removal and remediation with no current plans for reuse of structures.

For the containment building, the licensee considered two exposure scenarios to determine the most limiting for use in DCGL derivation: 1) industrial worker building inspection scenario and 2) building renovation/demolition scenario. For the industrial worker building inspection scenario, the worker is assumed to be exposed to penetrating radiation from sources, inhalation of airborne radioactive particulates, and inadvertent ingestion of radioactive material. The industrial worker is assumed to exhibit a more limiting occupancy factor since the final condition of the containment building will have no ventilation, lighting, or power. For the building renovation/demolition scenario, the licensee considered this scenario despite the fact that there are no plans to renovate or demolish the containment building after license termination as stated in section 6.6.5 of the LTP. The worker is assumed to be exposed to penetrating radiation from sources, inhalation of airborne radioactive particulates, and inadvertent ingestion of radioactive material. The scenarios, critical group determination, and exposure pathways are consistent with the guidance provided in NUREG-1757, Volume 2.

2.5.4 Mathematical Models

The licensee selected RESRAD Version 6.22 to derive site-specific soil DCGL values for the industrial worker scenario. The results from RESRAD for each radionuclide in units of mrem per pCi/g were scaled to the 25 mrem (0.25 mSv) TEDE limit to determine an acceptable DCGL value. Soil DCGLs were also evaluated for application to buried piping as stated in section 6.6.6 of the LTP. The licensee used an additional code, Microshield Version 5.05, to assess the potential dose from the buried piping.

For building surfaces and volumetrically contaminated bulk material, RESRAD-BUILD Version 3.22 was selected to derive site-specific DCGLs for the industrial worker scenario. The results from RESRAD-BUILD for each radionuclide in units of mrem per dpm/100 cm^2 were scaled to the 25 mrem (0.25 mSv) TEDE limit to determine an acceptable DCGL value. The licensee used an additional code, Microshield Version 5.05, to assess the annual dose rates from embedded piping.

After staff review, the mathematical models selected by the licensee are consistent with the conceptual model of the site. The computer codes used for derivation of DCGLs and dose assessment are acceptable.

2.5.4.1 Site-Specific Parameters for RESRAD

The licensee used the process outlined in Figure 6-5 of the LTP to select conservative values for input parameters that have great influence on radiation dose results. The selection process is consistent with the guidance provided in NUREG/CR-6676, -6692, -6767, -6755, and -6697.

In the first step of the selection process, the RESRAD input parameters were classified as one of the following three types: behavioral, metabolic, or physical, which is consistent with

NUREG/CR-6697. Behavioral parameters depend on the behavior of the receptor and the scenario definition. Metabolic parameters represent the metabolic characteristics of the receptor and are independent of the scenario definition. Physical parameters are those parameters that would not change if a different group of receptors was considered.

In the second step of the selection process, parameters were prioritized in order of their respective importance in dose modeling according to NUREG/CR-6697. Four attributes were considered in determining the priority of a parameter: (1) the relevance of the parameter in dose calculations, (2) the variability of the dose as a result of changes in the parameter value, (3) the parameter type, and (4) the availability of parameter-specific data. The parameters that have a large influence on dose results and are site-specific (i.e., physical) were assigned a higher priority than parameters that have small influence on dose results and/or are behavioral or metabolic parameters. Three levels of priority (1, 2, and 3) were used.

According to the priority, parameter type, availability of site-specific data, and relevance in dose calculation, a parameter was treated as either deterministic or stochastic in the third step. Deterministic parameters were assigned fixed values without further analysis. The licensee elected to treat the behavioral and metabolic parameters as deterministic. The values for behavioral and metabolic parameters were assigned values the RESRAD default library. Using the RESRAD default values should result in conservative dose estimates. Where site-specific data was available, the licensee treated the physical parameters deterministically. The remaining physical parameters for which no site-specific data were available, the licensee classified as either priority 1, 2, or 3. Priority 1 and 2 parameters were treated stochastically whereas priority 3 physical parameters were treated deterministically. The value for stochastic parameters were determined by their correlation with the resulting dose. Probabilistic sensitivity analysis that was incorporated into the RESRAD code was used to study the correlation.

Step four of the selection process involved using RESRAD to conduct a sensitivity analysis for the stochastic parameters. In the sensitivity analysis, each stochastic parameter was assigned a generic distribution obtained from NUREG/CR-6697, Attachment C or NUREG/CR-6767, Attachment C; while the deterministic parameters were assigned fixed values that were site-specific or RESRAD defaults. The partial rank correlation coefficient (PRCC) reported by RESRAD was used as the index to characterize a stochastic parameter's sensitivity. If the absolute value of PRCC was greater than 0.25, the parameter was characterized as sensitive; otherwise, the parameter was characterized as insensitive. The threshold value of 0.25 was consistent with the approach in NUREG/CR-6676.

After the sensitivity analysis was conducted, the last step of the selection process was to assign values to the input parameters. Physical parameters were assigned values according to the following rules:

1. Parameters for which site-specific data were available were assigned site-specific values.
2. Priority 1 and 2 parameters shown to be sensitive (with absolute PRCC values greater than 0.25) were assigned conservative values, either 75th (positive correlation) or 25th (negative correlation) quantile value of its generic distribution. If the 75th quantile value was less than the mean value of the distribution, the mean value was assigned to the parameter.

3. Priority 1 and 2 parameters shown to be insensitive (absolute values of PRCC less than 0.25) were assigned a median value from their generic distributions.
4. Priority 1 and 2 parameters shown to be insensitive but correlated with a parameter shown to be sensitive were assigned values based on the conservative value assigned to the sensitive parameter.
5. Priority 3 parameters were assigned values from the RESRAD default library.

The selection process takes into account the site-specific physical environment, sensitivities of parameters, and a receptor's behavioral pattern and metabolic conditions. The process used is consistent with NRC guidance and should result in derivation of conservative DCGLs.

2.5.4.2 Sensitivity Analysis

Probabilistic sensitivity analysis was used to study the influence of the input parameters on dose results, identify the important parameters, and assign parameter values, as discussed in the parameter selection process. The parameter distributions used in the sensitivity analysis are the generic distributions from NUREG/CR-6697; the actual distribution should be narrower because the generic distributions are based on national data. Determination of sensitive parameters was based on the values of the PRCC calculated by the RESRAD code for each individual parameter. The PRCC is a gauge of the correlation between the peak radiation dose and the parameter value. The larger the absolute value of the PRCC, the greater the influence of the parameter value on the peak dose. When the PRCC value is positive, the peak dose would increase with an increased parameter value. When the PRCC value is negative, the peak dose would decrease when the parameter value is increased. On the basis of previous studies (NUREG/CR-6755, -6692, and -6697) on uncertainty analysis of the RESRAD code, it was concluded by staff that the PRCC is the most representative among several coefficients of correlation between the peak dose and the parameter value.

The use of 75th or 25th quantile values for sensitive parameters in deterministic calculations would most likely generate conservative dose values, i.e. the peak dose would most likely be greater than the 75th quantile value of the corresponding peak dose distribution obtained from probabilistic calculations. Therefore, it is determined that the sensitivity analyses conducted for the LTP were comprehensive and the method used to select parameter values is acceptable and should result in conservative DCGLs.

2.5.4.3 Site-Specific Parameters for RESRAD-BUILD

For building surfaces and structures, the licensee used the process outlined in Figure 6.10 of the LTP to select conservative values for input parameters that have great influence on radiation dose results. The input parameter values were selected by using the parameter selection process described in section 2.5.4.1. The selection process takes into account the influence of the parameter values on the potential dose results. Based on the influence of the parameters, conservative values are assigned to the sensitive parameters to ensure calculation of conservative doses. The selection process was determined to be appropriate. The DCGLs based on the parameter selection process would be conservative.

The built-in capability of the RESRAD-BUILD code to conduct probabilistic sensitivity analyses was used to study the sensitivity of input parameters. The procedure used was the same as that discussed in section 2.5.4.1 for the RESRAD code except the threshold value of PRCC was

changed from 0.25 to 0.1. This choice of the PRCC value to identify sensitive parameters was consistent with NUREG/CR-6676 and was determined to be acceptable. The probabilistic sensitivity analysis considered the sensitivity of a parameter within its potential range in conjunction with the range of other parameters; therefore, it is considered to be more comprehensive and more appropriate than a deterministic sensitivity analysis when the range of the input parameter value is wide and the parameter value is quite uncertain. The choice of using 25th quantile, 75th quantile, or the mean value for sensitive parameters should result in conservative dose results. The sensitivity analysis in the LTP is determined to be appropriate and acceptable.

2.5.4.4 Comparison of the Peak Dose Results

The DCGL value for each radionuclide was derived on the basis of the peak dose of that radionuclide obtained from deterministic calculation. As mentioned in the previous section, it is most likely that the peak dose used to derive the DCGL value would be greater than 75th quantile value of the corresponding distribution obtained from probabilistic calculations. Staff has performed independent probabilistic calculations using the same parameter assignments as used in the licensee's sensitivity analysis and has confirmed the above expectation.

2.5.5 Derived Concentration Guideline Levels

2.5.5.1 Surface Soil DCGLs

The licensee has selected the DCGL approach to demonstrate compliance with 10 CFR 20.1402 for unrestricted release of the site. RESRAD Version 6.22 was used to determine DCGL values for the industrial worker scenario. Based on hydrogeological conditions, the licensee developed a mathematical model from the surface of the site to the groundwater zone. Details are provided in section 6.6.2.1 of the LTP. A spent fuel pool cooler pad soil sample was analyzed to identify the 26 potential radionuclides which are present at the site. The licensee believes that this sample has the highest level of contamination collected during the characterization process. Of the 26 radionuclides, only 6 radionuclides were identified in the soil sample. The radionuclides identified were C-14, Co-60, Ni-63, Sr-90, Cs-134, and Cs-137. Thus, soil DCGL values were derived for these six radionuclides.

Based on site conditions, the licensee selected the industrial worker scenario which reflects the realistic assumptions of site rather than using the conservative assumptions of the resident farmer scenario. One of the assumptions for this scenario included the suppression of the plant, meat, milk, and aquatic food pathways. The licensee believes that it is highly unlikely that gardening, farming, or production of aquatic foods from the Industrial Area will be established based on the justification provided in section 6.4.2.1. In the Industrial area, a worker would occupy the area for 2,000 hours per year (50-workweek year). The licensee assumed that the worker spends 50% of their time indoors and 50% outdoors while onsite. The drinking water pathway is not suppressed due to four potable water wells existing on the site.

Consistent with the guidance in NUREG-1757, Volume 2, section 3.3, the licensee considered the potential dose contribution from insignificant (discounted) radionuclides. The potential dose from these radionuclides was calculated using RESRAD Version 6.22. The total potential dose from the discounted radionuclides resulted in 0.572 mrem/year (5.72×10^{-3} mSv/year), based on a decay correction to correspond to an approximate FSS date of July 1, 2008.

The selection of parameter values for RESRAD calculations followed the process discussed in section 2.5.4.1. The final values for non-sensitive parameters and the assigned sensitive parameter values are listed in Appendix 6-A and Appendix 6-C of the LTP, respectively. The potential dose contribution from discounted radionuclides of 0.572 mrem/year (5.72×10^{-3} mSv/year) was subtracted from the regulatory dose criterion of 25 mrem/year (0.25 mSv/year), resulting in a reduced dose limit of approximately 24.4 mrem/year (0.244 mSv/year) for the site. Table 6-5 of the LTP lists DCGLs that will be used for residual radioactivity in soil. Staff reviewed the licensee's approach, performed independent analyses, and found it to be acceptable and consistent with the guidance in NUREG-1757, Volume 2.

2.5.5.2 Subsurface Soil DCGLs

In section 6.6.2.6 of the LTP, the licensee evaluated the applicability of surface DCGL values to subsurface soil contamination (i.e., soil contamination greater than 15 centimeters in soil depth) using the guidance in Appendix I of NUREG-1757, Volume 2. The licensee applied the unity rule to the mixture of the six detected radionuclides discussed in the above section and calculated a maximum radionuclide concentration using the soil DCGLs listed in Table 6-5 of the LTP. For mixed surface soil, the maximum radionuclide concentrations results in the 25 mrem/year (0.25 mSv/year) dose limit to the industrial worker. The licensee evaluated the dose effects from varying contamination depths ranging of 0.15, 0.5, 1.0, 1.5, 2.0, 2.5, and 3.0 meters. From 0.15 m to 0.5 m, the total peak of the mean dose increased by 9.05%, but there was little increase in total peak of the mean dose in increasing thicknesses up to 3 m as shown in Table 6-7 of the LTP. The licensee considered these results as non-conservative. The licensee determined that for areas up to 300 m^2 the non-conservatism would be covered by the area factors.

In addition, the licensee evaluated discrete pockets of contamination since subsurface soil contamination had been observed in discrete pockets onsite. The licensee modeled these discrete pockets as cylindrical soil volumes, 2 m deep with a surface area of 100 m^2. Peak of the mean doses were calculated with the pocket exposed to the surface at varying depths ranging from 0.25 m to 10 m. Results in Table 6-8 of the LTP demonstrate that the peak of the mean dose decreases with increasing depth. Based on these results, the licensee believes that using the surface soil DCGL values is more conservative than developing higher DCGL values for discrete pockets of subsurface soil contamination. In section 6.6.2.6.3, the LTP states the developed subsurface soil DCGL values would be non-conservative if the subsurface soil contamination is excavated later and spread on the surface, becoming surface soil contamination. Staff performed independent analyses to verify the licensee's results. Staff reviewed the methodology used and finds the approach acceptable.

2.5.5.3 Structural Surfaces DCGLs

The licensee selected RESRAD-BUILD Version 3.22 to derive the structural surface DCGL values for the suite of 26 radionuclides, based on an industrial worker building occupancy scenario, using the guidance provided in NUREG/CR-6755. RESRAD-BUILD can be used to estimate radiation exposure from both surface and volumetric sources. The mathematical model for derivation of structural surfaces DCGLs included four contaminated walls as well as a contaminated floor. The licensee did not include the ceiling in the modeling due to the following assumptions: 1) the ceiling was not contaminated, 2) the ceiling will be replaced if room would be reused, or 3) the ceiling would be an insignificant dose contribution to the receptor due to the large ceiling distance from the floor. The input parameters were selected using the process

described in section 2.5.4.3. The final list of parameter values used for DCGL derivation is found in Appendix 6-M. Table 6-9 of the LTP lists the structural surface DCGL values used for residual radioactivity that will remain on existing building surfaces. Staff independently performed the analysis using RESRAD-BUILD Version 3.3 to verify licensee's results. Staff found the approach to be acceptable and consistent with the guidance provided in NUREG/CR-6755 and NUREG/CR-6676.

2.5.5.4 Bulk Material DCGLs

In section 6.6.4 of the LTP, the licensee describes the derivation of volumetric (bulk material) DCGL values for the suite of 26 radionuclides. The licensee determined that volumetric DCGL values were needed since some structures may have been potentially contaminated from neutron activation. Volumetric contamination may also exist due to the migration of surface contamination. RESRAD-BUILD Version 3.22 was utilized to derive DCGLs for activated or volumetrically contaminated bulk material. The mathematical model included only the floor area as a source of contamination. The LTP stated that most interior concrete in the containment building will be removed. Only the carbon steel liner and concrete below the liner which were areas below the reactor vessel have a potential of being activated. The licensee also assumed that structures remaining on the floor have the highest possibility of contamination volumetrically due to radioactive spills. The floor was modeled as a 1 foot thick (0.3 m) volume source with the same surface area (137 m^2) used in the derivation of structural surface DCGL values. The licensee assumed that a 1 foot thick (0.3 m) volume source is approximately the maximum depth of activation or contamination based on the guidance in NUREG/CR-5884, Volume 2. Following the process for parameter selection described in section 2.5.4.3, DCGL values for bulk material were developed. Table 6-10 of the LTP lists the bulk material DCGL values. The final list of parameter values used for DCGL derivation is found in Appendix 6-N and Appendix 6-O. Staff independently performed analyses using RESRAD-BUILD Version 3.3 to verify licensee's results. Staff reviewed the approach used by the licensee and found the approach to be acceptable and consistent with the guidance provided in NUREG/CR-6755 and NUREG/CR-6676.

2.5.5.5 Containment Building Interior Surfaces DCGLs

The LTP, section 6.6.5, describes the methodology used to derive the containment building DCGLs. As stated in the LTP, there will be no light, ventilation, and power in the containment building. The majority of the interior concrete will be removed, leaving only the carbon steel liner plate. Therefore, the licensee determined that the industrial worker scenario used to derive the structural surface DCGLs is an unrealistic scenario for application to the interior surface of the containment building. The licensee developed two sets of DCGLs for the containment building to determine the most limiting scenario in this case: 1) industrial worker building inspection scenario and 2) a building renovation/demolition scenario.

2.5.5.5.1 Industrial Worker Building Inspection Scenario

Based on characterization samples, nine out of the 26 radionuclides in the site-specific suite were identified for this scenario. The license utilized RESRAD-BUILD Version 3.3 to model three rectangular compartments, which included a floor, four walls, and a ceiling in each compartment. In the mathematical model, the licensee used a single compartment to represent the three compartments and modeled them as a cylindrical source with cylindrical walls and a

domed ceiling. Section 6.6.5.1.2 of the LTP describes model assumptions used in the mathematical model. Input parameters used for DCGL derivation are listed in Appendix 6-P. Staff reviewed the modeling assumptions used for the containment building and found the assumptions acceptable. Independent analysis performed by the staff verified licensee's results. Table 6-11 of the LTP lists the DCGL values for the industrial worker building inspection scenario.

2.5.5.5.2 Building Renovation/Demolition Scenario

Containment building DCGL values for 26 radionuclides potential present at the site were derived based on the building renovation/demolition scenario. The LTP states that there are no plans to renovate or demolish the containment building after license termination. The licensee determined that this scenario should be analyzed to investigate the most limiting scenario for DCGL derivation. The licensee used the building renovation scenario as described in NUREG/CR-5512, Volume 1 as a basis for this scenario. Using RESRAD-BUILD Version 3.3, the licensee followed the process for parameter selection described in section 2.5.4.3 to develop the DCGL values for the containment building. Appendix 6-Q and Appendix 6-R list parameters used for DCGL development. Table 6-12 of the LTP lists the DCGL values for the renovation/demolition scenario.

2.5.5.5.3 Application of Containment Building DCGLs

By comparing the DCGL values contained in Tables 6-11 and 6-12 of the LTP, the licensee determined that the building renovation/demolition scenario was more limiting than the industrial worker building inspection scenario. Thus, the more limiting DCGLs should be applied to the containment building. However, in section 6.6.5.4 of the LTP, the licensee states that a more conservative approach will be imposed in that structural surface DCGLs derived in section 6.6.3 of the LTP will be applied to the reasonably accessible surfaces of the containment building. The renovation/demolition DCGLs will be applied to the containment building dome surfaces. Worker safety during remediation and FSS activities was considered in selecting the application of containment building DCGLs. Staff found the approach acceptable.

2.5.5.6 Buried Piping

As discussed in section 6.6.6 of the LTP, the licensee evaluated using soil DCGLs for application to buried piping. The licensee estimated a total length of 30,700 linear feet (9,357 m) is to remain onsite after license termination. The buried piping ranges from one inch (2.54 cm) to 108 inches (274 cm) in diameter at a soil depth of at least three feet (0.91 m). Using Microshield, the licensee determined that a gross DCGL value of 100,000 dpm/100 cm^2 would be applied to the interior piping surface to ensure that the 25 mrem/year (0.25 mSv/year) dose criterion would be met. The licensee assumed mean nuclide fractions of 0.17 for Co-60 and 0.83 for Cs-137 based on site characterization for soil. As stated in the LTP, the licensee assumed that the buried piping disintegrates upon license termination. As such, the disintegrated media is assumed to be subsurface soil and the media volume is assumed to be equal to the piping volume. Soil contamination is assumed to be uniformly mixed within the volume. Staff reviewed the licensee's approach and found it reasonable and acceptable.

2.5.5.7 Embedded Piping

A total length of 5,360 linear feet (1,637 m) of embedded piping was estimated to be left at that site after license termination. The embedded piping ranges from 0.75 inches (1.91 cm) to 18 inches (45.7 cm) in diameter, with depths between 9.0 (22.9 cm) to 30 inches (76.2 cm) beneath the concrete surface. To derive the DCGLs for embedded piping, the licensee assumed a scenario in which an industrial worker is exposed to residual radioactivity from: 1) a location within the concrete-encased piping and 2) contaminated surfaces of the building. The licensee considers the potential dose from embedded piping to be additive along with the potential dose to the worker from residual radioactivity from building surfaces. The LTP, section 6.6.7, states that the surface DCGLs will be reduced by the dose contribution from embedded piping to ensure compliance with the dose criterion.

Using Microshield, the licensee determined that a DCGL value of 100,000 dpm/100 cm^2 would be applied to the interior surface of the embedded piping to ensure that the dose criterion would be met. The licensee assumed mean nuclide fractions of 0.802 for Cs-137 and 0.161 for Co-60, based on piping system samples from different buildings on the site. The amount of concrete shielding assumed for model input included the minimum concrete covering above the largest diameter of embedded piping. The licensee assumed an occupancy factor of 2,000 hours per year for the industrial worker. Results of the analysis shown in Table 6-13 of the LTP, estimate annual dose rates ranging from 0.0002 mrem/year (2.0×10^{-6} mSv/year) to 0.19 mrem/year (1.9×10^{-3} mSv/year). Based on the licensee's model assumptions, staff was able to independently verify licensee's dose results.

In addition to the building occupancy scenario, the licensee investigated a scenario which involved an industrial worker being potentially exposed to residual radioactivity from embedded piping removal activities. Using the published dose factors along with the assumptions for a pipe cutting and removal scenario in NUREG-1640, Volume 1, for a DCGL value of 100,000 dpm/100 cm^2, the annual dose rates were estimated to be 4.0 mrem/year (0.04 mSv/year) for Cs-137 and 2.4 mrem/year for Co-60 (0.024 mSv/year) . Based on this potential dose contribution from embedded piping, the licensee has committed to grout the piping which has residual contamination above the adjusted NRC screening levels of 20,000 dpm/100 cm^2 (NUREG/CR-5512, Volume 3). Staff reviewed the approach for deriving DCGLs for embedded piping and found the approach reasonable and acceptable.

2.5.6 Elevated Measurement Comparison DCGLs

Area factors are needed for elevated measurement comparisons during scanning in Class 1 areas. The number of static measurements needs to be adjusted if the sensitivity of the scanning technique is not capable for detecting levels of residual radioactivity below the DCGLs. Area factors are also needed to identify small areas with elevated residual radioactivity that may require further investigation.

The licensee calculated area factors for surface soil and structural surfaces based on the industrial worker scenario. For surface soils, area factors for the industrial farmer scenario were computed by running the RESRAD computer code repeatedly with changing areas of contamination as well as the parameter describing the "length parallel to aquifer flow" which affects the area of contamination. Area factors were computed for six radionuclides detected in soil at the site. The area factors for these radionuclides are listed in Table 6-15 of the LTP.

RESRAD-BUILD Version 3.22 was used to compute area factors for the building occupancy scenario for four of the 26 site-specific radionuclides. A total of eight radionuclides were discounted because they were not present above analytical minimum detectable activity levels. Area factors were derived for the principal gamma emitting radionuclides which includes Co-60, Cs-134, and Cs-137. Area factors for Sr-90 was calculated only to demonstrate that the area factors were conservatively bounded by the area factors for the principle gamma emitting radionuclides. For calculating area factors for the building occupancy scenario, the area of the source was varied from 137 m^2 to 0.5 m^2. The area factors for the building occupancy scenario are listed in Table 6-18 of the LTP.

Staff independently verified area factors using RESRAD Version 6.22 for the industrial worker scenario and RESRAD-BUILD Version 3.3 for the building occupancy scenario and found no discrepancies. Maintaining consistency between the derivation of the base-case DCGLs and $DCGL_{EMC}$ values gives reasonable assurance that doses from exposure to smaller areas with elevated residual radioactivity would not exceed the dose limit. The area factors will be used for class 1 areas at the site.

2.5.7 Alternate Exposure Scenarios

For the Rancho Seco site, the industrial worker scenario was considered the mostly likely scenario for license termination. However, the licensee investigated the dose impacts from two alternative scenarios: 1) a resident farmer scenario and 2) a cattle grazing scenario by an offsite member of the public.

2.5.7.1 Resident Farmer Alternate Exposure Scenario

The licensee selected RESRAD Version 6.22 as the mathematical model to calculate the potential dose to a resident farmer. The residential farming scenario in general assumes light farming activities resulting in continuous exposure to residual radioactivity remaining at the site via multiple exposure pathways. Potential exposure pathways considered include direct external exposure from residual radioactivity in soil material, internal exposure from inhalation of airborne radionuclides, and internal exposure from ingestion of (1) plant foods grown in the soil with residual radioactivity and irrigated with contaminated water, (2) meat and milk from livestock fed with contaminated fodder and water, (3) drinking water from a contaminated well, (4) fish from a contaminated pond, and (5) soil with residual radioactivity.

Table 6-5 of the LTP lists the DCGL values for the six radionuclides found in soil. The licensee assumed that the soil was contaminated with the maximum allowable radionuclide concentrations listed in Table 6-6 of the LTP. The input parameter values were selected by using the parameter selection process described in Figure 6-5 of the LTP. The parameters selected along with the appropriate justifications are provided in Appendix 6-W and the statistical parameter distributions used are provided in Appendix 6-X. The mathematical model calculated the mean dose for the six radionuclides at specified times, up to 1000 years, after license termination. Table 6-19 lists the dose results based on a resident farmer scenario.

The potential dose from the remaining 20 radionuclides from the suite of 26 radionuclides was evaluated using the resident farmer scenario. Using RESRAD Version 6.22 as the mathematical model, the licensee calculated the mean dose probabilistically at specified times following license termination. The dose results are provided in Table 6-19.

As shown in Table 6-19 of the LTP, approximately 25 years following license termination, the calculated dose of 29 mrem/year (0.29 mSv/year) for a resident farmer exceeds the dose criterion. However, 30 years after license termination, the calculated dose for the resident farmer is less than the dose criterion. The licensee provides the following justifications for not considering the resident farmer scenario within 30 years following license termination: 1) no plans for public transfer of the site based on the justifications presented in section 6.4.2 of LTP, 2) Class B and Class C radioactive waste will be stored onsite under the existing 10 CFR Part 50 license for an indefinite time period awaiting permanent disposal, and 3) Greater than Class C waste will be stored in the Independent Spent Fuel Storage Installation (ISFSI) under a 10 CFR Part 72 license until transfer to permanent waste disposal. The licensee believes that 30 years after license termination, the calculated dose for a resident farmer is comparable to the calculated dose to an industrial worker. Staff reviewed the licensee's alternative resident farmer exposure scenario along with the justifications and found the comparison of dose impacts to the industrial scenario to be reasonable and acceptable.

2.5.7.2 Cattle Grazing Alternative Exposure Scenario

As discussed in section 6.8.3 of the LTP, portions of the site (open range areas with low probability of contamination) are leased to local ranchers for cattle grazing. However, the cattle are not precluded from grazing in potentially contaminated areas of the site in the future. The licensee analyzed the dose impact of maintaining an industrial worker scenario but allowing cattle grazing and the consumption of meat from the grazing cattle by an offsite member of the public. The licensee assumed that the soil was contaminated with the maximum allowable radionuclide concentrations listed in Table 6-6 of the LTP. Using RESRAD Version 6.22, the licensee created the cattle grazing scenario by modifying some of the exposure pathways for a resident farmer. Modifications included the following: 1) suppression of all pathways except for meat ingestion, 2) no irrigation, 3) no livestock water intake for meat, and 4) no grain for beef cattle feed. The maximum peak of the mean dose resulted in 5.13 mrem/year (0.0513 mSv/year). The licensee believes that this dose does not need to be accounted for in the industrial worker scenario because the offsite member of the public is different from the industrial worker. The licensee states that there is no impact between the two scenarios. Staff reviewed the licensee's approach and found the comparison of dose impacts of the cattle grazing scenario to the industrial scenario acceptable.

2.5.8 ALARA Determination

The licensee has provided a conservative dose model to meet the 25 mrem/yr limit. This approach, along with the findings in the license termination rule's GEIS (Ref. 26) results in the licensee demonstrating compliance with the ALARA requirements of 10 CFR 20.1402.

The staff has reviewed the information in the LTP for Rancho Seco according to Section B.6 of NUREG-1700. Based on this review the staff has determined that the licensee has conformed to 10 CFR 50.82(a)(9)(ii)(D), and that the FSS plan in the LTP provides assurance that residual radioactive contamination levels will meet the criteria specified in Part 20, for unrestricted use.

2.6 <u>Site End Use</u>

Section 50.82(a)(9)(ii)(E) requires a licensee to provide a description of the planned end use of the site if the licensee proposes to have its license terminated under restricted conditions. The

licensee has proposed to have its license terminated with no restrictions on the use of the site, under the provisions of 10 CFR 20.1402. Therefore, the licensee is not required to provide a description of the planned end use of the site.

The staff finds that the licensee has conformed to 10 CFR 50.82(a)(9)(ii)(E) and the description is therefore acceptable.

2.7 Cost Estimate

An updated site-specific estimate of the remaining decommissioning costs to terminate the license is required by 10 CFR 50.82(a)(9)(ii)(F). The staff reviewed Section 7, Update of Remaining Site-Specific Decommissioning Costs, of the Rancho Seco License Termination Plan, Revision 0, dated April 2006.

The estimate to complete radiological decommissioning is $138.3 million. SMUD will place $27 million annually through 2008 into its external fund to provide full funding of the decommissioning project.

The staff reviewed the Rancho Seco LTP against the guidance of NUREG-1713, "Standard Review Plan for Decommissioning Cost Estimates for Nuclear Power Reactors." Based on this review, the staff determined that the licensee has met the requirements of 10 CFR 50.82(a)(9)(ii)(F) by providing an updated site-specific cost estimate for the remaining decommissioning activities and that the cost estimate is acceptable.

2.8 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 Rancho Seco's 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 Rancho Seco's decommissioning activities 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. In the 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 reviewed the information in the LTP for Rancho Seco, 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 part 51.53.

2.9 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 Rancho Seco 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 implementaion. Changes to the LTP not requiring NRC approval will be submitted as an update to the final safety analysis report, in accordance with 10 CFR 50.71e.

The staff concludes that authorizing the licensee to make certain changes, during the final site remediation, is acceptable, subject to the above listed conditions.

3.0 STATE CONSULTATION

In accordance with NRC regulations, the State of California was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATIONS

This amendment incorporates the Rancho Seco LTP and the LTP change process, which allows the licensee to make changes to the plan without NRC review and approval. Pursuant to 10 CFR 51.21, 51.32, and 51.35, an EA and Finding of No Significant Impact were published in the *Federal Register* on November 8, 2007.

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

5.0 CONCLUSIONS

The Commission 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 Commission'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.

6.0 LIST OF CONTRIBUTORS

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7.0 LIST OF ACRONYMS

ALARA	As Low As Is Reasonable Achievable
bgs	below ground surface
Bq/g	Becquerel per gram
Bq/L	Becquerel per liter
CFR	Code of Federal Regulations
CFS	Containment Foundation Sump
CCS	Continuing Characterization Survey
DCGL	Derived Concentration Guideline Limit
DOE	U.S. Department of Energy
dpm/100cm^2	disintegrations per minute per 100 square centimeters
DQO	Data Quality Objective
EA	Environmental Assessment
FR	Federal Register
FSME	Office of Federal and State Materials and Environmental Management Programs
FSS	Final Status Survey
GEIS	Generic Environmental Impact Statement
HSA	Historical Site Assessment
HTD	Hard to Detect
IA	Industrial Area
ICS	Initial Characterization Survey
ISFSI	Independent Spent Fuel Storage Installation
kV	kilovolt
LBGR	Lower Boundary of the Gray Region
LTP	License Termination Plan
MARSSIM	Multi-Agency Radiation Survey And Site Investigation Manual
MDC	Minimum Detectable Concentration
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
NRC	Nuclear Regulatory Commission
PAB	Primary Auxiliary Building
pCi/g	picocurie per gram
pCi/L	picocurie per Liter
PRCC	partial rank correlation coefficient
QA	Quality Assurance
QC	Quality Control
RAI	Request for Additional Information
RCA	Radiologically Controlled Area
RCRA	Resource Conservation and Recovery Act
REMP	Radiological Environmental Monitoring Program
RWST	Refueling Water Storage Tank
SCC	Secondary Component Cooling
SER	Safety Evaluation Report
SMUD	Sacramento Municipal Utility District
SNPS	Shoreham Nuclear Power Station
SRP	Standard Review Plan
Sv/hr	Sievert per hour

TEDE	Total Effective Dose Equivalent
TLG	TLG Services
TRU	Transuranic
uR/hr	microroentgen per hour

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