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October 30, 2014

DPG 14-233

U.S. Nuclear Regulatory Commission
Attn.: Document Control Desk
Washington, DC 20555

Docket No. 50-312

Rancho Seco Nuclear Station

License No. DPR-54

BIENNIAL UPDATE TO RANCHO SECO LICENSE TERMINATION PLAN

Attention: John Hickman

The Rancho Seco License Termination Plan (LTP), Revision 1 was submitted by the Sacramento Municipal Utility District (SMUD) under cover letter dated July 10, 2008. In accordance with 10 CFR 50.71(e), SMUD has updated the LTP to Revision 2 to reflect the current site conditions.

The enclosed attachments include removal/insertion instructions for the changed pages, a List of Effective Pages (included in the Table of Contents section), and the affected LTP pages. Vertical lines in the left hand margin of the affected pages indicate the area of changed text.

If you or members of your staff have questions, or require additional information or clarification, please contact me at einar.ronningen@smud.org or (916) 732-4817.

Sincerely,

Einar T. Ronningen
Superintendent, Rancho Seco Assets

ER/BG
Enclosures
Cc: NRC Region IV

NM$\$$501

Sacramento Municipal Utility District

Rancho Seco Nuclear Generating Station

License Termination Plan

Revision 2

October 2014

Rancho Seco LTP Revision 2 Replacement Pages

The listing below provides the replacement pages for Revision 2 of the LTP. The listing below only identifies the LTP pages that result in changes to page numbers from Revision 1 of the 2008 LTP. The replacement pages are based on double-sided pages. All changes to Revision 1 can be found in the Revision Section that precedes the Table of Contents of the LTP.

Cover:	Remove Revision 1 cover page and replace with Revision 2 cover page.
TOC:	Remove Revision 1 TOC replace with Revision 2 TOC (includes Inside Cover page and pages i through viii).
Chapter 1:	Remove all of Revision 0 Chapter 1 including Chapter TOC. Replace with Revision 2 Chapter 1 including the Chapter TOC (pages 1-i, 1-ii).
Chapter 2:	None
Chapter 3:	Remove Revision 0 TOC. Replace with Revision 2 TOC (pages 3-i, 3-ii). Remove Revision 0 pages 3-1 and 3-2, and replace with Revision 2 page 3-1 and Revision 0 page 3-2. Remove Revision 1 pages 3-9 and 3-10. Replace with Revision 1 page 3-9 and Revision 2 page 3-10.
Chapter 4:	None
Chapter 5:	Remove Revision 1 pages 5-41 and 5-42, and replace with Revision 2 pages 5-41 and Revision 1 page 5-42. Remove Revision 1 pages 5-45 and 5-46, and replace with Revision 2 page 5-45 and Revision 1 page 5-46. Remove Revision 1 pages 5-55 through 5-60, and replace with Revision 2 pages 5-55 through 5-59 and Revision 1 page 60.
Chapter 6:	Remove Revision 1 pages 6-33 through 6-36, and replace with Revision 2 pages 6-33 through 6-36.
Chapter 7:	Remove Revision 0 TOC and replace with Revision 2 TOC (pages 7-i, 7-ii). Remove Revision 0 pages 7-3 through 7-8, and replace with Revision 2 pages 7-3 through 7-7 and Revision 0 page 7-8.
Chapter 8:	None

Rancho Seco Nuclear Generating Station

License Termination Plan

SMUD
Sacramento Municipal Utility District

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APPENDICES

List of Revised Sections

Chapter	Section	Page	Revision
1	1.3.1	1-2	Update/Clarification regarding decommissioning of SSC's important to safety
1	1.3.2	1.3	Identify presence of part 50 licensed facility with the industrial area
1	1.4.1	1-4	Identify Waste Control Specialists as approved disposal site
1	1.4.1	1-4	Minor correction / clarification
1	1.4.2	1-5	Update current status of waste disposal/decommissioning activities
1	1.5.3	1-6	Identify Waste Control Specialists as approved disposal site
1	1.5.7	1-8	Update decommissioning fund information
3	3.1	3-1	Identify Waste Control Specialists as approved disposal site, update decommissioning schedule
3	3.3.6.1	3-10	Update waste disposal status
3	3.3.6.2	3-10	Update decommissioning schedule
5	5.4.3.2	5-41	Minor correction / clarification re: instrument calibration services
5	5.4.3.4.6	5-45	Minor correction / clarification re: Gamma Spec services
5	5.8.2.1.1	5-55	Update current site organization and responsibilities
5	5.8.2.1.2	5-56	Update current site organization and responsibilities
6	6.6.7	6-33	Minor correction / clarification re: embedded pipe decommissioning
7	7.1.1	7-3	Update site/ waste disposal/ decommissioning status

LIST OF EFFECTIVE PAGES

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1.0 GENERAL INFORMATION

The Sacramento Municipal Utility District (the District) is submitting this License Termination Plan (LTP) for Rancho Seco Nuclear Generating Station (Rancho Seco). The following provides the licensee name, address, license number, and docket number for Rancho Seco:

Sacramento Municipal Utility District
Rancho Seco Nuclear Generating Station
14440 Twin Cities Road
Herald, CA 95638
License No. DPR-54
Docket No. 50-312

All of the Rancho Seco spent nuclear fuel is stored in the:

Rancho Seco Independent Spent Fuel Storage Installation (ISFSI)
14440 Twin Cites Road
Herald, CA 95638
License No. SNM-2510
Docket No. 72-11

1.1. Purpose

The objective of decommissioning Rancho Seco is to reduce the level of residual radioactivity to levels that permit the release of the site for unrestricted use and allow for the termination of the 10 CFR Part 50 license. The Rancho Seco LTP satisfies the requirement in 10 CFR 50.82(a)(9) to submit an LTP for Nuclear Regulatory Commission (NRC) approval. The LTP is a supplement to the Rancho Seco Defueled Safety Analysis Report (DSAR) and is accompanied by a proposed license amendment that establishes the criteria for when changes to the LTP require prior NRC approval.

1.2. Scope

The District prepared the LTP using the guidance in:

- Regulatory Guide 1.179 "Standard Format and Contents for License Termination Plans for Nuclear Power Reactors," [Reference 1-1],
- NUREG-1575 "Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)," [Reference 1-2],
- NUREG-1700 "Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans," [Reference 1-3], and
- NUREG-1757 "Consolidated NMSS Decommissioning Guidance." [Reference 1-4]

The LTP includes a discussion on the following:

- Site Characterization to ensure that final status surveys (FSS) cover all areas where contamination existed, remains, or has the potential to exist or remain,
- Identification of remaining dismantlement activities

- Plans for site remediation,
- A description of the FSS plan to confirm that the plant and site will meet the release criteria in 10 CFR Part 20, Appendix E,
- Dose modeling scenarios that ensure compliance with the radiological criteria for license termination,
- An estimate of the remaining site-specific decommissioning costs, and
- A supplement to the environmental report describing any new information or significant environmental change, since the submittal of the Supplement to Rancho Seco Environmental Report - Post Operating License Stage, associated with proposed license termination activities.

Section 1.5 discusses the purpose and content of each LTP chapter. Section 1.6 discusses the process for making changes to the LTP.

1.3. Historical Background and Site Description

1.3.1 Historical Background

Rancho Seco was a 913-MWe pressurized water reactor (PWR) designed by Babcock and Wilcox Company. The District shut down Rancho Seco permanently on June 7, 1989 after approximately 15 years of operation. On August 29, 1989, the District formally notified the NRC that the plant was shut down permanently.

On May 20, 1991, the District submitted the Rancho Seco Decommissioning Plan and on March 20, 1995, the NRC issued an Order approving the Decommissioning Plan and authorizing the decommissioning of Rancho Seco. In March 1997, the District submitted its Post Shutdown Decommissioning Activities Report (PSDAR), in accordance with 10 CFR 50.82. The PSDAR superseded the original Decommissioning Plan and provided the information required by 10 CFR 50.82(a)(4).

The District began actively decommissioning Rancho Seco in February 1997. The transfer of all of the spent nuclear fuel to the 10 CFR Part 72 ISFSI on August 21, 2002 and the Greater Than Class C (GTCC) waste generated during decommissioning was transferred to the ISFSI on August 22, 2006. Accordingly, the only quality-related structures, systems, or components (SSCs) at the Rancho Seco 10 CFR Part 50 licensed site were the radioactive sources used to calibrate the instrumentation used to measure radioactivity in gaseous and liquid effluents. Plant dismantlement is substantially complete and all of the SSCs that *were* safety-related or important-to-safety have been removed from the plant and shipped for disposal.

1.3.2 Site Description

The Rancho Seco site is located in the southeast part of Sacramento County, California. It lies either wholly or partly within Sections 27, 28, 29, 32, 33, and 34 of Township 6 North, Range 8E. The site is approximately 26 miles north-northeast of Stockton and 25 miles southeast of Sacramento. The Rancho Seco nuclear reactor unit and ISFSI lie wholly within Section 29.

More generally, the site is located between the Sierra Nevadas to the east and the Coast Range along the Pacific Ocean to the west in an area of flat to lightly rolling terrain at an elevation of approximately 200 feet above mean sea level. To the east of the site the land becomes more rolling, rising to an elevation of 600 feet at a distance of about seven miles, and increasing in elevation thereafter approaching the Sierra Nevada foothills.

The area surrounding the site is almost exclusively agricultural and is presently used as grazing land and more recently for growing grapes. The climatology of the Rancho Seco site is typical of the Great Central Valley of California. Cloudless skies prevail during summer and much of the spring and fall. The rainy season usually extends from December through March.

The owner-controlled site is approximately 2,480 acres with all acreage being owned by the District. Within the owner-controlled area is an approximately 87-acre fence-enclosed Industrial Area containing the nuclear facility. A 30-acre natural gas-fired power plant is located approximately ½ mile south of the Industrial Area boundary. Also within the 2,480-acre site are:

- The 560-acre Rancho Seco Reservoir and Recreation Area,
- A 50-acre solar power (photo-voltaic) electrical generating station,
- The 10-acre, 10 CFR Part 72 licensed ISFSI[1], and
- An emergency backup data center (located within the Industrial Area) used to recover critical computer applications and data if a serious incident or disaster disables data servers at District headquarters in Sacramento.
- The District back up control center (located within the Industrial Area), which is used to control the District's electrical system in the event that the control facility at District headquarters needs to be evacuated.
- An approximately one acre fenced boundary enclosing the Interim On-Site Storage Building (IOSB), which comprises the remaining part 50 licensed activities.

Groundwater in the site area occurs under free or semi-confined conditions. Groundwater movement in the area is to the southwest with a slope of about ten feet/mile.

There is no indication of faulting beneath the site. The nearest fault system, the Foothill Fault System, is about ten miles east of the site and has been inactive since the Jurassic Period, some 135 million years ago. Ground accelerations of no greater than 0.05g are anticipated at the site.

The soils at the Rancho Seco site can be categorized as hard to very hard silts and silty clays with dense to very dense sands and gravels.

1.4. Decommissioning Approach

1.4.1 Overview

The objective of decommissioning Rancho Seco is to reduce the level of residual radioactivity to levels that permit the use of the site for unrestricted use and allow for the termination of the 10 CFR Part 50 license. Decommissioning involves the systematic removal of SSCs that

[1] The 10 CFR Part 72 licensed ISFSI is independent of the 10 CFR Part 50 licensed facility.

comprise the radioactive portions of the site. The District conducts decommissioning activities in accordance with the NRC's Decommissioning Rule, the Rancho Seco 10 CFR Part 50 license, plant Licensing Basis Documents, and approved procedures.

After SSCs are removed, they are surveyed to determine the contamination level. Non-contaminated material is free-released for asset recovery, recycling, or disposal at an offsite landfill. Contaminated material may be released as non-contaminated material after decontamination, shipped to a licensed offsite processor for disposition, or shipped to an offsite low-level waste (LLW) disposal site (i.e., Energy*Solutions*[2]or Waste Control Specialists, Inc[3]).

Radioactive waste handlers package LLW for transport and disposal in accordance with applicable NRC and Department of Transportation (DOT) regulatory requirements.

Rancho Seco continues to implement its Radiological Controls Program. The objectives of the Radiological Controls Program are to control radiation hazards, avoid accidental radiation exposures, maintain worker Total Effective Dose Equivalent (TEDE) to less than 5 rem/year, and maintain doses to workers and the public As Low As Reasonably Achievable (ALARA). The philosophies, policies, and objectives of the Radiological Controls Program are based on federal regulations and associated regulatory guidance.

The Rancho Seco ALARA program is implemented in accordance with the requirements of 10 CFR Part 20 and additional NRC regulatory guidance. The ALARA policy states management's commitment to maintain exposures to workers and the public ALARA. This commitment is contained in the DSAR and is implemented by plant administrative procedures and Radiation Protection Department implementing procedures.

The integrated approach to decommissioning includes support from the Radiation Protection, Quality Assurance, Engineering, Maintenance, Licensing, and Decommissioning organizations and outside contractors, as required to complete the project. The Decommissioning organization provides project management and has developed administrative procedures to implement decommissioning activities. Additionally, staff uses existing plant programs and procedures to implement various aspects of the decommissioning project.

The use of trained individuals, adherence to approved procedures and established institutional controls, will ensure that the risk to the public and worker health and safety is minimal. Risks associated with the transportation of LLW are also minimal.

The environmental assessment, discussed in Chapter 8 of this LTP, determined that the environmental effects from decommissioning of Rancho Seco are minimal, and there are no adverse effects outside the bounds of NUREG-0586 "Final Generic Environmental Impact Statement (GEIS) on Decommissioning of Nuclear Facilities," [Reference 1-5]. Additionally the conclusions contained in the Supplement to Rancho Seco Environmental Report - Post Operating License Stage, used as the original basis for the environmental assessment of radiological and non-radiological effects of decommissioning, are still valid.

The District's dose modeling objective is to develop Derived Concentration Guideline Levels (DCGLs) that will demonstrate compliance with the dose-based release criteria. The District will then demonstrate through the FSS that the levels of residual radioactivity at the site are

[2] Energy*Solutions* was previously Envirocare of Utah, [3] WCS approved for disposal of class B & C waste in 2013

equal to or below the DCGLs (i.e., below the dose-based release criteria) with a pre-specified degree of confidence.

1.4.2 Approach to License Termination

The District intends to release the Rancho Seco site for unrestricted use in two phases, with the license terminated after completion of the second phase. The first phase includes the majority of the site, including impacted and non-impacted areas, except for the Interim Onsite Storage Building (IOSB). In general, each location will be released after the completion of the associated final status surveys. Once an area has been verified as ready for release, no additional surveys or decontamination of the subject area will be required unless the controls (e.g., administrative or engineered) established to prevent re-contamination have been compromised.

Following completion of an FSS for a given survey unit, Rancho Seco staff will develop an FSS Report to document the final radiological condition of the area and demonstrate that the criteria in 10 CFR 20.1402 are met. These reports will be compiled and submitted to the NRC. Following the completion and acceptance of the FSS Reports for the first phase, the District will submit a license amendment request to release the first portion of the site for unrestricted use.

The disposal facility operated by Waste Control Specialists, Inc was determined to be an acceptable disposal site for class B & C radioactive waste by SMUD in 2013. These Class B & C radioactive wastes are being shipped for disposal, with an anticipated completion in the fourth quarter of 2014. Once the class B & C wastes are removed from the IOSB phase II of the decommissioning will commence including FSS for the IOSB and the submittal of a license to release the remainder of the site and terminate the 10 CFR Part 50 license in 2016.

Chapter 5 of this LTP, Final Status Survey Plan, describes the contents of the FSS Report.

The spent nuclear fuel and the GTCC waste will remain in storage at the ISFSI until the Department of Energy (DOE) transfers this waste to a federal repository.

1.5. Plan Summary

1.5.1 General Information

The Rancho Seco LTP describes the process used to meet the requirements for terminating the Rancho Seco 10 CFR Part 50 license and release the site for unrestricted use. The LTP has been prepared in accordance with the requirements in 10 CFR 50.82(a)(9) and is submitted as a supplement to the Rancho Seco DSAR. The LTP submittal is accompanied by a proposed license amendment that establishes the criteria for when changes to the LTP require prior NRC approval.

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

1.5.2 Site Characterization

LTP Chapter 2 discusses the site characterization that has been conducted to determine the extent and range of radioactive contamination on site prior to remediation, including remaining structures, soils, and surface and ground water at Rancho Seco. Based on the results of the site characterization, Rancho Seco staff will plan remediation and FSSs in impacted areas.

The District also used the information gathered during site characterization to develop site-specific input into the dose modeling.

The Historical Site Assessment (HSA) provided the foundation for further site characterization. The HSA provided the preliminary information required to divide the site into survey units. The survey units were evaluated against the criteria specified in the MARSSIM guidelines for classification. Data from subsequent characterization may be used to change the original classification of an area, within the requirements of this LTP, up to the time of the FSS, as long as the classification reflects the level of residual activity existing prior to any remediation in the area.

1.5.3 Identification of Remaining Site Dismantlement Activities

LTP Chapter 3 identifies the remaining site dismantlement and decontamination activities. The information provided in Chapter 3 includes:

- A description of the areas and equipment that need further remediation,
- A characterization of radiological conditions that may be encountered,
- Estimates of associated occupational radiation dose,
- An estimate of the types and quantities of radioactive material to be released in accordance with 10 CFR 20.2001, and
- A description of proposed control mechanisms to ensure areas are not re-contaminated.

The District is decommissioning Rancho Seco in accordance with the DECON alternative described in NUREG-0586. Completion of the DECON option is contingent upon access to a LLW disposal site. Until 2013, Rancho Seco's access to acceptable disposal facilities consisted of Energy*Solutions*' Clive, Utah facility, which accepts only Class A radioactive waste. Waste Control Specialists' Andrews , Texas facility was determined to be an acceptable disposal site for class B & C radioactive waste by SMUD in 2013. These Class B & C radioactive wastes are being shipped to WCS for disposal, with anticipated completion in the fourth quarter of 2014.

Decommissioning activities are conducted in accordance with the Rancho Seco Radiation Protection Program, Radwaste Manual, Off-Site Dose Calculation Manual (ODCM), Safety Program, and plant administrative procedures. These are established programs that are routinely inspected by the NRC.

Activities conducted during decommissioning do not pose any greater radiological or safety risk than those conducted during plant operations. The radiological risk associated with decommissioning activities is bounded by previously analyzed radiological risk for former operating activities that occurred during major maintenance and outage evolutions.

The information contained in Chapter 3 supports the assessment of impacts considered in other sections of the LTP and provides sufficient detail to identify resources needed during the remaining dismantlement activities.

1.5.4 Site Remediation Plans

LTP Chapter 4 discusses the various remediation techniques that may be used during decommissioning to reduce residual contamination to levels that comply with the release criteria in 10 CFR 20.1402. LTP Chapter 4 also discusses the ALARA evaluation and the Radiation Protection Program requirements that will be implemented during the remediation process.

The remediation method used is dependent on the contaminated material. The principal materials that may be subjected to remediation are structural surfaces and soils. LTP Appendix 4-A describes the equipment, personnel, and waste costs used to generate a unit cost basis for the various remediation actions that may be used.

Following the removal of equipment and components, structures will be surveyed and decontaminated, as necessary. Remediation techniques that may be used for 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. Operational constraints and dust control will be addressed in site excavation and soil control procedures. 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 for 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 implemented during decommissioning is similar to the program that was in place during commercial power operation. Decommissioning does not present any new challenge to the Radiation Protection Program and the existing program is adequate to safely control the radiological aspects of remediation activities.

1.5.5 Final Status Survey Plan

LTP Chapter 5 discusses the Final Status Survey Plan. The FSS Plan has been prepared using applicable regulatory and industry guidance. This plan will be used to develop site procedures and work instructions to perform the FSS of the Rancho Seco site.

The FSS Plan describes the final survey process used to demonstrate that the Rancho Seco facility and site complies with radiological criteria for unrestricted use specified in 10 CFR 20.1402 (i.e., annual dose limit of 25 mrem plus ALARA for all dose pathways). NRC regulations applicable to radiation surveys are found in 10 CFR 50.82(a)(9)(ii)(D) and 10 CFR 20.1501(a) and (b).

The FSS Plan describes the development of the survey plan, survey design and data quality objectives, survey methods and instrumentation, data collection and processing, data assessment and compliance, and the Quality Assurance Project Plan (QAPP). This FSS Plan address only facilities and land areas that are identified as contaminated or potentially contaminated (impacted) resulting from activities associated with commercial nuclear plant operation.

As discussed above, the District intends to release the site in two phases. The first phase includes the majority of the site and remaining structures. The second phase of site release includes the IOSB following the disposal of Class B and C radioactive waste. The FSS Plan addresses requirements applicable to the first phase of site release and may also be used during the second phase to release the IOSB.

The ISFSI, licensed under 10 CFR Part 72, is not subject to the conditions of this LTP.

1.5.6 Compliance with the Radiological Criteria for License Termination

LTP Chapter 6, along with Chapters 4 & 5, describes the methods used to demonstrate compliance with the radiological criteria for license termination and release of the site for unrestricted use. Chapter 6 discusses the site-specific inventory of radionuclides, future land use scenarios, exposure pathways, computational models used for dose modeling, sensitivity analysis, DCGLs, the derivation of area factors, and a comparison of alternative exposure scenarios for impacted area soils.

The District intends on maintaining ownership of the 2,480 acre Rancho Seco site. Accordingly, dose modeling is based on the Industrial Worker scenario. Chapter 6 provides justification for using this scenario.

1.5.7 Update of Site-Specific Decommissioning Costs

LTP Chapter 7 provides an estimate of the remaining decommissioning costs for releasing the site for unrestricted use. This chapter also compares the estimated remaining cost with the funds currently available in the decommissioning trust fund.

The final trust fund contribution was made in 2008. Currently the fund contains approximately $31 million, which is estimated to be sufficient to complete the remaining decommissioning activities.

1.5.8 Supplement to the Environmental Report

LTP Chapter 8 updates the environmental report for Rancho Seco with new information and any significant environmental impacts associated with the site's decommissioning and license termination activities. This section of the LTP is prepared pursuant to 10 CFR 51.53(d) and 10 CFR 50.82(a)(9)(ii)(G).

In accordance with 10 CFR 51.53(d), the District submitted the Supplement to Rancho Seco Environmental Report - Post Operating License Stage along with the original Decommissioning Plan in 1991. This Environmental Report addressed the actual or potential environmental impacts associated with Custodial and Hardened-SAFSTOR, and provided an initial assessment of the effects of Deferred-DECON.

The Supplement to Rancho Seco Environmental Report - Post Operating License Stage compared Rancho Seco decommissioning attributes to those identified in NUREG-0586. NUREG-0586 provides a generic environmental assessment of decommissioning a reference nuclear facility. When the NRC issued the Decommissioning Rule in 1988, and based on the findings in the GEIS, it concluded a generic finding of "no significant (environmental) impact." The NRC further concluded that no additional Environmental Impact Statement would need to be prepared in connection with the decommissioning of a particular nuclear site unless the impacts of a particular plant have site-specific considerations significantly different from those studied generically. The Supplement to Rancho Seco Environmental Report - Post Operating License Stage concludes that Rancho Seco falls within the envelope of the GEIS.

Additionally, in accordance with the California Environmental Quality Act (CEQA), the District conducted an initial study of the potential environmental impacts resulting from closing and decommissioning Rancho Seco. Based on the results of that study, the District staff prepared a Negative Declaration stating that decommissioning would not have a significant environmental impact.

In February 1997, the District began dismantlement activities at Rancho Seco, with the goal of terminating the 10 CFR Part 50 license by 2008. Prior to beginning dismantlement activities, the District conducted another evaluation under CEQA and again concluded that decommissioning would not have a significant environmental impact.

In March 1997 the District submitted its PSDAR, in accordance with 10 CFR 50.82. The PSDAR superseded the original Decommissioning Plan and provided the information required by 10 CFR 50.82(a)(4). PSDAR Section 4, "Environmental Review," provides a discussion of the environmental impacts associated with site-specific decommissioning activities and concluded that all of the decommissioning attributes identified for Rancho Seco are within the envelop of NUREG-0586, except for the decommissioning cost estimate, which is not directly comparable.

The environmental assessment determined that the environmental effects for decommissioning Rancho Seco are minimal, and there are no adverse effects outside the bounds of NUREG-0586 or the associated Supplement 1. Additionally, the conclusions contained in the Supplement to Rancho Seco Environmental Report - Post Operating License Stage, used as the original basis for the decommissioning environmental assessment of radiological and non-radiological effects of decommissioning, are still valid.

1.6. License Termination Plan Change Process

The District is submitting the LTP as a supplement to the DSAR. Accordingly, the District will update the LTP in accordance with 10 CFR 50.71(e). Once approved, the District may make changes to the LTP, without prior NRC approval, in accordance with the criteria in 10 CFR 50.59, 10 CFR 50.82(a)(6), and 10 CFR 50.82(a)(7).

The District also submitted a proposed amendment to the Rancho Seco Operating License that adds a license condition that establishes the criteria for determining when changes to the LTP require prior NRC approval. Changes to the LTP require prior NRC approval when the change:

- Increases the probability of making a Type I decision error above the level stated in the LTP,
- Increases the radionuclide-specific DCGLs,

- Increases the radioactivity level, relative to the applicable DCGL, at which investigation occurs, and
- Changes the statistical test applied other than the Sign Test or Wilcoxon Rank Sum Test.

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

1.7. License Termination Plan Information Contact

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rjones2@smud.org

1.8. References

1-1 U.S. Nuclear Regulatory Commission Regulatory Guide 1.179 "Standard Format and Contents for License Termination Plans for Nuclear Power Reactors"

1-2 U.S. Nuclear Regulatory Commission NUREG-1575 "Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)"

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

1-4 U.S. Nuclear Regulatory Commission NUREG-1757 "Consolidated NMSS Decommissioning Guidance"

1-5 U.S. Nuclear Regulatory Commission NUREG-0586 "Final Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities."

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TABLES

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3.0 IDENTIFICATION OF REMAINING DECOMMISSIONING ACTIVITIES

3.1 Introduction

In accordance with 10 CFR 50.82 (a)(9)(ii)(B), the License Termination Plan (LTP) must identify the major remaining dismantlement and decontamination activities. This chapter was written following the guidance of NUREG-1700, "Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans," [Reference 3-1] and Regulatory Guide 1.179, "Standard Format and Content of License Termination Plans for Nuclear Power Reactors," [Reference 3-2] and will discuss those dismantlement activities as of December 31, 2005. Information is presented to demonstrate that these activities will be performed in accordance with 10 CFR 50 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). Information that demonstrates that these activities will not have a significant effect on the quality of the environment is provided in LTP Chapter 8, Supplement to the Environmental Report.

The information includes those areas and equipment 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.

Sacramento Municipal Utility District's (District's) primary goals are to decommission the Rancho Seco Nuclear Generating Station (Rancho Seco) safely and to maintain the continued safe storage of spent fuel in an Independent Spent Fuel Storage Installation (ISFSI). The District will decontaminate and dismantle Rancho Seco in accordance with the DECON alternative, as described in NUREG-0586, "Final Generic Environmental Impact Statement" (FGEIS) [Reference 3-3]. Completion of the DECON option is contingent upon access to one or more low-level waste (LLW) disposal sites. Currently, Rancho Seco has access to the disposal facilities of Energy*Solutions*'[1] Clive, Utah facility, which accepts only Class A radioactive waste. Waste Control Specialists' Andrews, Texas facility was determined to be an acceptable disposal site for class B & C radioactive waste by SMUD in 2013. Completion of the second phase of site release will be after these Class B & C radioactive wastes are shipped to WCS for disposal, schedule for completion in the fourth quarter of 2014..

The District is currently conducting decontamination and dismantlement (D&D) activities at the Rancho Seco site in accordance with the Rancho Seco Post Shutdown Decommissioning Activities Report (PSDAR) [Reference 3-4]. Decommissioning activities are being coordinated with the appropriate Federal and State regulatory agencies in accordance with plant administrative procedures. All special nuclear material (spent fuel) is located at the ISFSI. By the end of the second quarter of 2006, it is expected that all greater than Class C (GTCC) waste material will also be located at the ISFSI.

Decommissioning activities at Rancho Seco are conducted in accordance with the Rancho Seco Defueled Safety Analysis Report (DSAR) [Reference 3-5], Permanently Defueled Technical Specifications [Reference 3-6], Rancho Seco Quality Assurance Program (QAP) [Reference 3-7], existing 10 CFR Part 50 license, and the requirements of 10 CFR 50.82(a)(6) and (a)(7). If an activity requires prior Nuclear Regulatory Commission (NRC) approval under 10 CFR 50.59(c)(2) or a change to the Rancho Seco Permanently Defueled Technical Specifications or license, a submittal shall be made to the NRC for review and approval before implementation of the activity in question.

[1] Energy*Solutions* was previously Envirocare of Utah

Decommissioning activities are conducted in accordance with the Rancho Seco Radiation Protection Program, the Off-Site Dose Calculation Manual (ODCM), Safety Program, and the Radwaste Manual. Such activities are and shall be conducted in accordance with these established programs that are frequently inspected by the NRC. Activities conducted during decommissioning do not pose any greater radiological or safety risk than those conducted during former plant operations. Decommissioning activity radiological risk is bounded by previously analyzed radiological risk for former operating activities that occurred during major maintenance and outage evolutions.

The activities described in Section 3.3, Future Decommissioning Activities, include activities up to the future release of the site. This section provides 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 D&D tasks is also provided. This information includes 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. This information supports the assessment of impacts considered in other sections of the LTP and provides sufficient detail to identify inspection or technical resources needed during the remaining dismantlement activities. Many of these dismantlement tasks require coordination with other federal, state or local regulatory agencies or groups.

The dismantlement activities described in Section 3.3 provide the NRC the information to support site release and future license termination pursuant to 10 CFR 50.82(a)(11)(i). Therefore, this section was written to clearly indicate each dismantlement activity that remains to be completed prior to qualifying for license termination. The final state of the Industrial Area will be a partially abandoned facility (as defined in Chapter 1 of this LTP) 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 the District for industrial purposes.

3.2 Completed Decommissioning Activities and Tasks

3.2.1 Spent Fuel Storage

The District signed the contract in 1992 for the design, licensing and fabrication of a transportable storage system. In 1995 the ISFSI was constructed and fabrication of the cask and associated equipment began. However, in 1996, quality issues throughout the dry storage industry and vendor bankruptcy forced work to be stopped. In 1997, a new supplier resumed the design and license work.

The transportable storage system consists of a transportation cask, twenty-one dry storage canisters, twenty-two horizontal storage modules and a multi-axle trailer. The cask serves for on-site transfer and off-site transportation overpack for the canisters. The canisters hold the spent fuel in a structural array and are then seal-welded. The horizontal storage modules are thick reinforced concrete storage bunkers used to store the canisters. The twenty-second module will provide storage for GTCC waste from reactor vessel internals.

3.3.3.1 Reactor Building

Some liner decontamination is expected to be required once internal structures are removed. Cleaning, up to and including paint removal, will be done as necessary to meet the derived concentration guideline level (DCGL).

3.3.3.2 Auxiliary Building

Extensive decontamination is planned for rooms below grade level in the Auxiliary Building. Many of the rooms were exposed to leaking or spraying water systems and decontamination is expected to include extensive surface removal including core boring and sawing. It may be necessary to remove floors or sumps if contamination extends through the concrete. These rooms are currently undergoing removal of obstacles that will interfere with 100% scanning surveys.

3.3.3.3 Spent Fuel Pool

Significant decontamination is required for the Spent Fuel Pool. The pool liner has been removed and one interior wall where significant pool liner leakage has occurred is scheduled for removal. Once the wall is removed decontamination of remaining wall and floor surfaces will occur as well as the cleaning of the embedded leak chases and through-wall pipes.

3.3.3.4 Turbine Building

The Turbine Building has only minor contamination levels with little decontamination planned with the exception of selected floor drain piping segments and sumps.

3.3.3.5 Embedded Pipe Systems

Embedded pipe systems are located in all of the impacted buildings listed above. Most embedded system piping is for floor drains. Cleaning is in progress with an initial high-pressure wash to remove debris followed by an abrasive grit blast process as required. Once cleaned to acceptable limits most embedded piping will be grouted to mitigate reuse or transport of remaining residual activity.

3.3.3.6 Wastewater Systems

While most wastewater piping that will remain is believed to be below DCGL levels, the Retention Basins and associated bottom drains will require remediation and/or partial removal. The radioactive discharge line from the RHUTs to the Retention Basins is expected to be removed. Storm drains that lead directly offsite and storm drains that collect system drainage and lead to the outfall should require no remediation. Some system piping that leads to the storm drains is currently being removed in lieu of extensive surveys. Cleaning is currently underway on the oil/water separator. Oil and sludge will be removed and a FSS will be performed.

3.3.4 Non-Radiological Activities

3.3.4.1 Outbuilding Demolition

The demolition of temporary outbuildings continues. The remaining concrete pads will be surveyed as a part of the FSS process.

3.3.4.2 Site Grading

Once Phase I site release is obtained, low areas will be filled and graded for drainage. These areas include the cooling tower basins and canal, the spray ponds and the below grade portion of the Turbine Building including the circulating water lines. Other grading and landscaping may occur.

3.3.5 Control Mechanisms to Ensure No Recontamination

Due to the large scope of remaining structures and systems to be decontaminated and the need for some FSS activities to be performed in parallel with dismantlement activities, a systematic approach to controlling areas is established. Upon commencement of the FSS for survey areas where there is a potential for re-contamination, isolation and control measures will be implemented as described in Section 5.2.4.4 of this LTP.

3.3.6 Deferred Activities

3.3.6.1 Storage of Class B and C Waste

In 2013, SMUD management made the decision that Waste Control Specialist's facility in Andrews Texas provided an acceptable waste disposal option for the stored class B and C waste. As a result, the waste will be shipped for disposal in 2014 and the building will be decontaminated as required.

3.3.6.2 Final Status Survey of IOSB

Once the IOSB is decontaminated a FSS will be performed in accordance with this LTP and a final release from the Part 50 license will be requested. The time frame for that request is currently scheduled for 2017.

3.4 Radiological Impacts of Decontamination and Dismantlement Activities

3.4.1 Occupational Exposure

Figure 3-1 provides Rancho Seco cumulative site dose and estimates for the decommissioning project. These estimates were developed to provide site management ALARA goals. The goals are verified by summation of actual site dose, as determined by appropriate dosimetry. ALARA estimates are a compilation of work plan (radiation work permit) estimates for the period. This information is in addition to information gathered for reporting of yearly site dose in accordance with the Rancho Seco Quality Manual (RSQM), Appendix A. The annual report of occupational dose meets the guidance of NRC Regulatory Guide 1.16, "Reporting of Operating Information-Appendix A Technical Specifications," [Reference 3-8]. The total nuclear worker exposure during decommissioning is currently estimated to be less than 200 person-rem. This

5.4.3.2 Calibration And Maintenance

Instruments and detectors are calibrated for the radiation types and energies of interest at the site. The calibration source for beta survey instruments is typically Cs-137 because the average beta energy (188 keV) approximates the beta energy of the radionuclides found on surfaces or in piping on site (average beta energy of 166 keV). The alpha calibration source when used is typically Pu-239 that has an appropriate alpha energy for plant-specific alpha emitting nuclides. Gamma scintillation detectors are typically calibrated using Cs-137. Actual sources used for calibration will be determined by the licensed calibration facility performing the calibrations.

Instrumentation used for final status survey will be calibrated and maintained in accordance with the contract calibration facility program and procedures. Radioactive sources used for calibration are traceable to the National Institute of Standards and Technology (NIST) and have been obtained in standard geometries to match the type of samples being counted. When characterized HPGe detectors are used, suitable NIST-traceable sources are used for calibration, and the software is set up appropriately for the desired geometry. As vendor services are being used, these will be obtained in accordance with purchasing requirements for quality related services, to ensure the same level of quality.

5.4.3.3 Response Checks

Instrumentation response checks are conducted to assure proper instrument response and operation. An acceptable response for field instrumentation is an instrument reading within ±20% of the established check source value. Laboratory instrumentation standards will be within ±3 sigma as documented on a control chart. Response checks are performed daily before instrument use and again at the end of use. Check sources contain the same type of radiation as that being measured in the field and are held in fixed geometry jigs for reproducibility. If an instrument fails a response check, it is labeled with a Rancho Seco "Radiac Repair Tag" and is removed from service until the problem is corrected in accordance with applicable procedures. Measurements made between the last acceptable check and the failed check are evaluated to determine if they should remain in the data set.

5.4.3.4 Minimum Detectable Concentration (MDC)

The MDC is determined for the instruments and techniques used for final status surveys (Table 5-12). The MDC is the concentration of radioactivity that an instrument can be expected to detect 95 percent of the time.

5.4.3.4.1 Static MDC For Structure Surfaces

For static (direct) surface measurements, with conventional detectors, such as those listed in Table 5-12, the MDC is calculated by Equation 5-7 as follows:

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

Equation 5-7

where:

MDC_{static} = minimum detectable concentration for direct counting (dpm/100 cm^2),

B = number of background counts during the count interval t,

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

K = calibration constant (counts/min per dpm/100 cm^2).

The value of K includes correction factors for efficiency (ε_i and ε_s). The value of ε_s is dependent on the material type. Corrections for radionuclide absorption have been made.

5.4.3.4.2 Structural Surface Beta-Gamma Scan MDCs

Following the guidance of Sections 6.7 and 6.8 of NUREG-1507, MDCs for surface scans of structural surfaces for beta and gamma emitters will be computed by Equation 5-8 below. For determining scan MDCs, a rate of 95% of correct detections is required and a rate of 60% of false positives is determined to be acceptable: therefore, a sensitivity index value of 1.38 was selected from Table 6.1 of NUREG-1507 and Equation 5-7 becomes:

$$MDC_{structuralsurfacescan}(dpm/100cm^2) = \frac{1.38\sqrt{B}}{\sqrt{p}\,\varepsilon_i\,\varepsilon_s\left(\frac{A}{100}\right)t}$$

Equation 5-8

where:

B = number of background counts during the count interval t,

p = surveyor efficiency,

ε_i = instrument efficiency for the emitted radiation (cpm per dpm),

ε_s = source efficiency (intensity) in emissions per disintegration,

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

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

The numerator in Equation 5-8 represents the minimum detectable count rate that the observer would "see" at the performance level represented by the sensitivity index. The surveyor efficiency (p) will be taken to be 0.5, as recommended by Section 6.7.1 of NUREG-1507. The factor of 100 corrects for probe areas that are not 100 cm^2. In the case of a scan measurement, the counting interval is the time the probe is actually over the source of radioactivity. This time depends on scan speed, the size of the source, and the fraction of the detector's sensitive area that passes over the source; with the latter depending on the direction of probe travel. The source efficiency term (ε_s) in Equation 5-8 may be adjusted to account for effects such as self-absorption, as appropriate.

5.4.3.4.3 Total Efficiency (ε_t) and Source Efficiency (ε_s) for Concrete Contamination

The source term inventory on contaminated concrete appears to be primarily located within the top ten millimeters of the concrete surface. Various fixed point measurement alternatives for determining the source term were evaluated including gross beta measurements on the surfaces, volumetric concrete sampling and *in situ* gamma spectroscopy. Gross beta fixed point measurements were determined to be cost-effective and technically defensible under the

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

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

Equation 5-11

where:

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

p = Surveyor efficiency = 0.5.

A corresponding minimum detectable exposure rate can be determined for a specified detector and radionuclide by dividing the $MDCR_{surveyor}$ value by the detector manufacturer's count rate to exposure rate ratio (cpm per µR/h) to give a minimum detectable exposure rate in units of µR/h. The minimum detectable exposure rate is then used to determine the minimum detectable radionuclide concentration (i.e., the Scan MDC) by modeling a specified small area of elevated activity using MicroShield™ to yield a conversion factor of µR/h per pCi/g. The minimum detectable exposure rate is then divided by the MicroShield™ conversion factor to give a Scan MDC in units of pCi/g.

5.4.3.4.6 HPGe Spectrometer Analysis

The onsite chemistry laboratory no longer exists: therefore, gamma isotopic spectrometers that are calibrated to various sample geometries, including a one-liter marinelli geometry for soil analysis are provided by off-site contract lab vendor services. These systems are calibrated using NIST-traceable mixed gamma sources. Laboratory counting systems have software controlled count times which are set to meet a maximum MDC of 0.15 pCi/g for Cs-137 in soil; this is calculated by Equation 5-12 as follows:

$$MDC\ (pCi/g) = \frac{3 + 4.65\sqrt{B}}{K * V * t}$$

Equation 5-12

where:

B = number of background counts during the count interval t,

K = proportionality constant that relates the detector response to the activity level in a sample for a given set of measurement conditions,

V = mass of sample (g), and

t = count time (minutes)

In the event that HPGe detectors are obtained for *in situ* gamma spectroscopy of soils and structures. Their sensitivity will be similar to that of the lab spectrometer and is documented in DTBD-06-003.

5.4.3.4.7 Pipe Survey Instrumentation

Remaining pipe will be surveyed to ensure residual remaining activity is less than the DCGL. Pipe survey instruments proposed for use with pipe having diameters between 0.75 and 18 inches have been shown to have efficiencies ranging from approximately 0.02 to 0.51 (Table 5-12). This equates to detection sensitivities of approximately 350 dpm/100 cm^2 to 5,200 dpm/100 cm^2. This level of sensitivity is adequate to detect residual activity below the embedded pipe DCGL of 100,000 dpm/100 cm^2.

5.5 Data Collection and Processing

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

5.5.1 Sample Handling and Record Keeping

Sample collection and handling requirements are provided for each sample from the point of collection through obtaining the final results to ensure the validity of the sample data. Sample tracking records are controlled and maintained and, upon completion of the data cycle, are transferred to Document Control, in accordance with applicable procedures.

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

5.5.2 Data Management

Survey data are collected from several sources during the data life cycle and are evaluated for validity throughout the survey process. QC replicate measurements are not used as final status survey data (See Section 5.8.2.4.1 for design and use of QC measurements.). Measurements performed during turnover and investigation surveys can be used as final status survey data if they were performed according to the same requirements as the final status survey data. These requirements are:

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

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

3) The quality of the data collected is adequate,
4) All phases of package design and survey are properly reviewed, with QC and management oversight provided, and
5) Corrective actions, when identified, are implemented in a timely manner and are determined to be effective.

The following sections describe the basic elements of the FSS QAPP.

5.8.2.1 Project Management and Organization

An FSS organization will be established for the Rancho Seco site in RSAP-1901. This organization will be responsible for planning and implementation of final status surveys. Since the FSS organization has not been fully implemented at the time of LTP development, specific job titles may vary over the period of project execution. However, the following descriptions refer to various functional areas of responsibility and do not necessarily correspond to specific job titles. It is also important to note qualified individuals may assume the responsibilities of more than one of the functional positions described below. The FSS organization consists of the following functional areas.

5.8.2.1.1 Dismantlement Project Manager

The Dismantlement Project Manager, under the direction of the Superintendent, Rancho Seco Assets, has overall responsibility for program direction, technical content, and ensuring the program complies with applicable NRC regulations and guidance. This supervisor is responsible for preparation and implementation of the FSS procedures. Additional responsibility areas may include resolution of issues or concerns raised by the NRC or other Stakeholders, as well as programmatic issues raised by Rancho Seco site management. The Dismantlement Program Manager provides overall FSS project coordination, which may include, but is not limited to, interfaces with site personnel in areas of nuclear licensing, demolition and waste disposal. This individual is also tasked with ensuring the appropriate interface between various site functional groups is specified in work order documents and possess specific knowledge regarding Radiation Protection, FSS, and Industrial Safety requirements.

5.8.2.1.2 Final Status Survey Technical Specialist

Responsibilities of FSS Technical Specialist(s) 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 review of survey packages and data collected in support of the Final Status Surveys. FSS Technical Specialists are also responsible for control and implementation of survey packages during field activities. Specific responsibilities are likely to include:

- Coordination of turnover surveys,
- Survey area preparation (e.g., gridding),

- Ensuring final status survey sampling is conducted in accordance with applicable procedures and work instructions,
- Maintaining access controls over completed FSS survey areas,
- Determining survey area accessibility requirements,
- Coordination and scheduling of FSS Technicians to support the decommissioning schedule, and
- Ensuring all necessary instrumentation and other equipment is available to support survey activities.

The Final Status Survey Technical Specialist(s) is responsible for maintaining the FSS data records in both electronic formats and hardcopy files, as applicable. This includes maintaining survey measurement data and supporting data files and generating reports of survey results. Responsibilities also include maintaining the integrity of the FSS database and implementing FSS Database QA requirements.

5.8.2.1.3 Final Status Survey Technician

Final Status Survey Technicians are responsible for performance of final status survey measurements and collection of final status survey samples in accordance with applicable site procedures and survey package instructions. An FSS Technician will be responsible for maintaining the pedigree of instrumentation used in the survey by implementing the procedural requirements for calibration, maintenance and daily checks. Final Status Survey Technicians will be trained and task-qualified for the performance of the final status activities assigned to them. Final Status Survey Technicians may also participate in survey area preparations.

5.8.2.2 Written Procedures

Sampling and survey tasks must be performed properly and consistently in order to assure the quality of final status survey results. The measurements will be performed in accordance with approved, written procedures. Approved procedures describe the methods and techniques used for final status survey measurements. Those procedures have been cited in Section 5.9.1.

5.8.2.3 Training and Qualification

Personnel performing final status survey measurements will be trained and qualified. Training will include the following topics:

- Procedures governing the conduct of the FSS,
- Operation of field and laboratory instrumentation used in the FSS, and
- Collection of final status survey measurements and samples.

Qualification is obtained upon satisfactory demonstration of proficiency in implementation of procedural requirements. The extent of training and qualification will be commensurate with the education, experience and proficiency of the individual and the scope, complexity and nature of the activity required to be performed by that individual. Records of training and qualification will be maintained in accordance with approved training procedures

5.8.2.4 Measurement and Data Acquisitions

The FSS records have been designated as quality documents and will be governed by site quality programs and procedures. Generation, handling and storage of the original final status survey design and data packages will be controlled by site procedures. Each final status survey measurement will be identified by individual, date, instrument, location, type of measurement, and mode of operation.

5.8.2.4.1 Quality Control Surveys

Procedures establish built-in Quality Control checks in the survey process for both field and laboratory measurements, as described in LTP Section 5.8.2.2. For structures and systems, QC replicate scan measurements will consist of resurveys of a minimum of 5% of randomly selected class 1, 2, or 3 survey units typically performed by a different technician with results compared to the original survey result. The acceptance criterion shall be that the same conclusion as the original survey was reached based on the repeat scan. If the acceptance criterion is not met, an investigation will be conducted to determine the cause and corrective action.

Quality Control for direct surface contamination and/or gamma direct measurements will consist of repeat measurements of a minimum of 5% of the survey units using the same instrument type, taken by a different technician (except in cases where there is only one instrument or specialized training is required to operate the equipment) and the results compared to the original measurements using the same instrument type. The acceptance criterion for direct measurements is specified in approved procedures.

For soil, water and sediment samples, Quality Control will consist of participation in the laboratory Inter-comparison Program. However, as an additional quality measure, approximately 5% of such samples may be subjected to blind duplicate samples and/or third party analyses. The acceptance criterion for blank samples is that no plant-derived radionuclides are detected. The criterion for blind duplicates is that the two measurements are within the value specified by approved procedure. For third party analyses, the acceptance criterion is the same as those for blind duplicates. Some sample media, such as asphalt, will not be subjected to split or blind duplicate analyses due to the lack of homogeneity. These samples will simply be recounted to determine if the two counts are within 20% of each other, when necessary.

If QC replicate measurements or sample analyses fall outside of their acceptance criteria, a documented investigation will be performed in accordance with approved procedures; and if necessary, the Corrective Action Process described in Section 5.8.3.3 will be implemented. The investigation will typically involve verification that the proper data sets were compared, the relevant instruments were operating properly and the survey/sample points were properly identified and located. Relevant personnel are interviewed, as appropriate, to determine if proper instructions and procedures were followed and proper measurement and handling techniques were used including chain of custody, where applicable. When deemed appropriate, additional measurements are taken. Following the investigation, a documented determination is made regarding the usability of the survey data and if the impact of the discrepancy adversely affects the decision on the radiological status of the survey unit.

5.8.2.4.2 Instrumentation Selection, Calibration and Operation

Proper selection and use of instrumentation will ensure that sensitivities are sufficient to detect radionuclides at the minimum detection capabilities as specified in Section 5.4.3.4 as well as assure the validity of the survey data. Instrument calibration will be performed with NIST traceable sources using approved procedures. Issuance, control and operation of the survey instruments will be conducted in accordance with the Instrumentation Program procedures.

5.8.2.5 Chain of Custody

Responsibility for custody of samples from the point of collection through the determination of the final survey results is established by procedure. When custody is transferred outside of the organization, a chain of custody form will accompany the sample for tracking purposes. Secure storage will be provided for archived samples.

5.8.2.6 Control of Consumables

In order to ensure the quality of data obtained from FSS surveys and samples, new sample containers will be used for each sample taken. Tools used to collect samples will be cleaned to remove contamination prior to taking additional samples. Tools will be decontaminated after each sample collection and surveyed for contamination.

5.8.2.7 Control of Vendor-Supplied Services

Vendor-supplied services, such as instrument calibration and laboratory sample analysis, will be procured from appropriate vendors in accordance with approved quality and procurement procedures.

5.8.2.8 Database Control

Software used for data reduction, storage or evaluation will be fully documented. The software will be tested prior to use by an appropriate test data set.

5.8.2.9 Data Management

Survey data control from the time of collection through evaluation is specified by procedure. Manual data entries will be secondarily verified.

5.8.3 Assessment and Oversight

5.8.3.1 Assessments

QC will perform assessments of FSS activities in accordance with the Quality Assurance Program. The findings will be tracked and trended in accordance with existing procedures.

5.8.3.2 Independent Review of Survey Results

Randomly selected survey packages (approximately 5%) from survey units will be independently reviewed by the Quality Assurance personnel to ensure that the survey measurements have been taken and documented in accordance with approved procedures.

5.8.3.3 Corrective Action Process

The corrective action process, already established as part of the site's 10 CFR Part 50 Appendix B Quality Assurance Program, will be applied to FSS for the documentation, evaluation, and implementation of corrective actions. The process will be conducted in accordance with approved procedures which describe the methods used to initiate potential deviation from quality (PDQ) reports and resolve self assessment and corrective action issues related to FSS. The PDQ evaluation effort is commensurate with the classification of the PDQ and could include root cause determination, extent of condition reviews, and preventive and remedial actions.

5.8.3.4 Reports to Management

Reports of audits and trend data will be reported to management in accordance with approved procedures.

5.8.4 Data Validation and Verification

Survey data will be reviewed prior to evaluation or analysis for completeness and for the presence of outliers. Comparisons to investigation levels will be made and measurements exceeding the investigation levels will be evaluated. Procedurally verified data will be subjected to the Sign test, the Unity Sign test, the WRS test, or WRS Unity test as appropriate. Technical evaluations or calculations used to support the development of DCGLs will be independently verified to ensure correctness of the method and the quality of data.

5.8.5 Confirmatory Measurements

The NRC may take confirmatory measurements to make a determination in accordance with 10 CFR 50.82(a)(11) that the FSS and associated documentation demonstrate the site is suitable for release in accordance with the criteria for decommissioning in 10 CFR Part 20, subpart E. Confirmatory measurements may include collecting radiological measurements for the purpose of confirming and verifying compliance with NRC standards for unrestricted license termination. Timely and frequent communications with the NRC will ensure it is afforded sufficient opportunity for these confirmatory measurements prior to implementing any irreversible decommissioning actions.

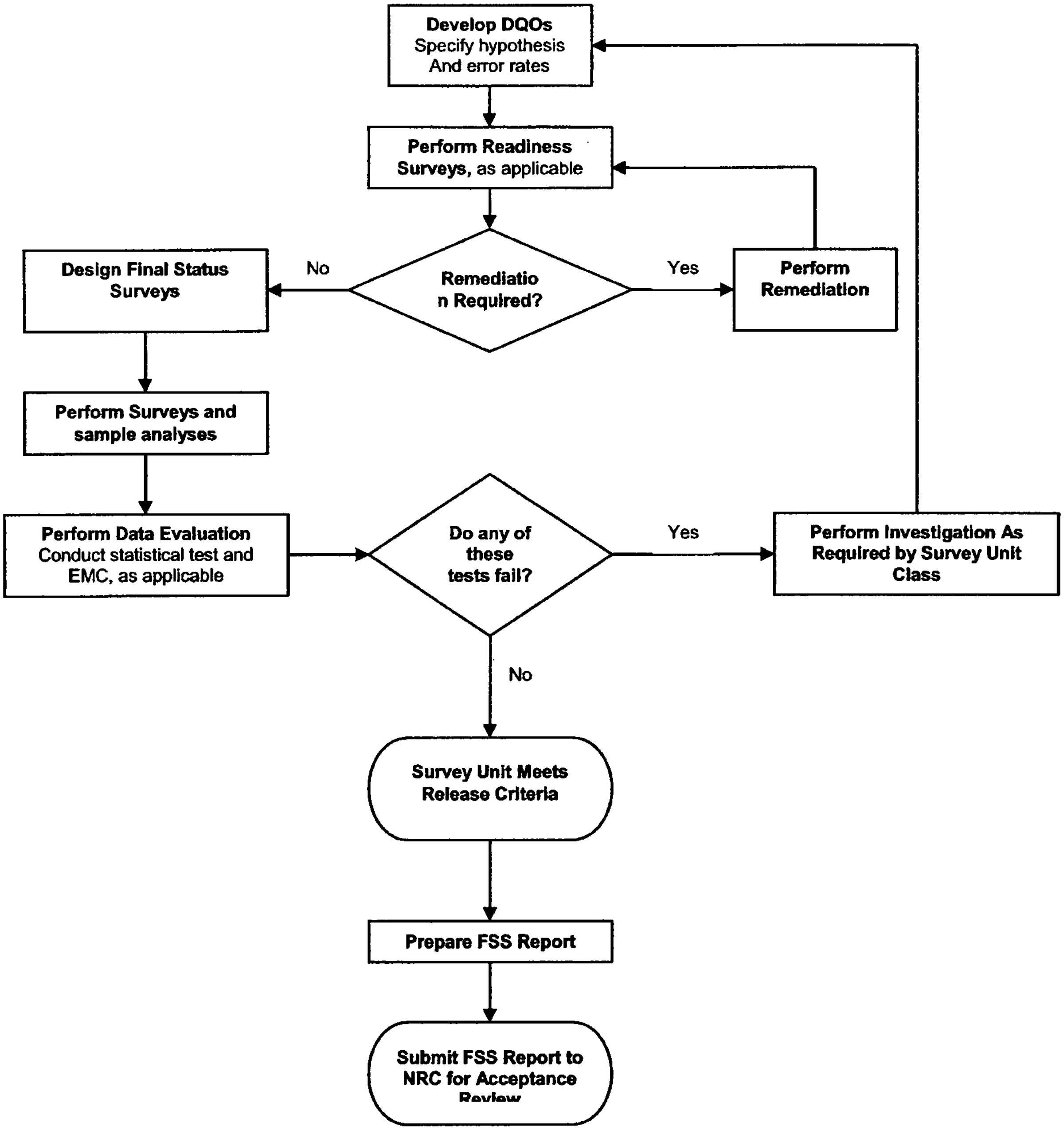

Figure 5-1

FSS Process Overview

6.6.6 Buried Piping

Approximately 30,700 linear feet of buried pipe have been identified that is expected to remain at Rancho Seco after license termination. The buried pipe ranges from one inch I.D. to 108 inch I.D. and is associated with systems such as the nitrogen gas system (one inch I.D.) to the main circulating water system (108 inch I.D.). Buried piping that will remain following license termination is located at a soil depth of three or more feet. A majority of the buried piping that is associated with systems that contained known contamination has been excavated during decommissioning and piping systems remaining have a low potential for significant internal contamination.

Evaluation of the buried piping scenario utilized soil DCGL values derived in Section 6.6.2. Under the scenario, buried piping, contaminated on the interior surface, is assumed to disintegrate instantaneously upon license termination. The disintegrated media is assumed to be subsurface soil and the media volume is assumed to be equal to the piping volume with the contamination uniformly mixed in the soil volume. A gross DCGL value to apply to interior piping surface was derived using standard computational methods assuming the disintegrated media is contaminated to soil DCGL concentrations using average observed nuclide fractions for soil and piping surface contamination.

The calculations assumed an average radionuclide mixture of 0.17 for Co-60 and 0.83 for Cs-137 (95% C. L.). A conservative gross DCGL of 100,000 dpm/100 cm^2 on the interior surface of the buried pipe was found acceptable based upon these calculations. The details of this analysis were developed in Rancho Seco DTBD-05-013, "Buried Piping Scenario and DCGL Determination Basis," [Reference 6-26].

6.6.7 Embedded Piping

Approximately 5,360 linear feet of embedded pipe have been identified that will remain at Rancho Seco. The embedded pipe ranges from 0.75 inch I.D. to 18 inch I.D. and is associated with the Turbine Building, Auxiliary Building, Reactor Building, Fuel Building, and IOSB drains. Embedded pipe is located at the drain entrance down to depths between 9 to 30 inches or more beneath the concrete surface, depending on the building. Only the IOSB embedded pipe remains to be decommissioned. This piping consists of less than 300 feet of primarily 4 inch drain pipe.

The embedded piping scenario assumes that the piping remains in place following decommissioning and that the dose to the industrial worker is from direct gamma exposure from the residual activity in the pipe with allowance made for photon attenuation by the wall or floor thickness of concrete remaining over the pipe. Whole body dose from the embedded pipe will be considered additive along with the dose to the industrial worker resulting from residual activity on the walls or floors of the room or area in which the embedded pipe is present. The surface DCGL will be reduced by the dose contribution from the embedded piping in order to ensure compliance with the annual dose limit.

Embedded pipe is partially shielded and constrained by the encasing concrete that limits the dose to the industrial worker to that arising from the gamma emitters in the nuclide mixture. The impact of nuclides that are not gamma emitters is minimal because the pipe is not easily extracted nor is the interior surface readily accessible through the overlying concrete. A total of 53 samples were collected and analyzed by gamma spectroscopy from various drains, sumps, and trenches in the buildings previously mentioned. Twenty samples were selected that reflect the different piping systems covered by the 53 samples. In many instances, several samples were collected from one system. The radionuclide analyses indicated that the primary contributors to whole body dose are Cs-137 and Co-60. The Fuel Building pipe sample results indicate the presence of a small portion of non-gamma emitters in the nuclide fraction. The mean nuclide fractions for Cs-137 and Co-60, as determined by the 20 samples, were 0.802 and 0.161 respectively. The individual building mean fractions were within two standard deviations of the overall mean values indicating a consistent nuclide ratio. This compares well with the concrete nuclide fractions of 0.89 and 0.11 for Cs-137 and Co-60 respectively.

A conservative gross DCGL of 100,000 dpm/100 cm^2 on the interior surface of the embedded pipe was evaluated and found acceptable. MicroShield® runs were used to model the gamma exposure at one meter from the concrete surface resulting from 100,000 dpm/100 cm^2 (4.5E-4 $\mu Ci/cm^2$) in the maximum size pipe in a given building one meter from the surface of the concrete covering the embedded pipe. The amount of the concrete shielding included in the model was based on the thinnest concrete covering above the largest diameter embedded pipe for the given building as determined from site drawings. An occupancy factor of 2,000 hours per year was assumed to calculate the annual dose rate. Results are shown in Table 6-13 below. The annual dose rates are all less than 1 percent of the 25 mrem/y annual limit. The details of this analysis were developed in Rancho Seco DTBD-05-009, "Embedded Piping Scenario and DCGL Determination Basis," [Reference 6-27].

Table 6-13

Embedded Pipe Annual Dose Rate By Building

Building	Turbine	Fuel	Auxiliary	Reactor
Max Pipe Size (inches)	8	8	6	18
Concrete Depth (inches)	18	30	9	12
Annual Dose Rate (mrem/y)	0.01	0.0002	0.19	0.12

The potential for the removal of the embedded pipe and consequent dose to an industrial worker at some time in the future was examined even though this was not part of the industrial worker building occupancy scenario. The published source of dose factors that came the closest to a pipe cutting and removal scenario was NUREG-1640, Volume 1, "Radiological Assessments for Clearance of Materials from Nuclear Facilities," [Reference 6-28]. If the mean dose factors (NUREG-1640, Volume 1, Table 3.24) and scenario for converting pipe into scrap material as outlined in NUREG-1640 are employed using a DCGL of 100,000 dpm/100 cm^2 and the given nuclide fraction for embedded pipe, the annual dose rates are calculated to be 4.0 mrem/y for Cs-137 and 2.7 mrem/y for Co-60. The dose contribution from Cs-137 was principally from the release of contamination and subsequent inhalation by the worker while the dose from Co-60 was mostly the whole body dose from handling the contaminated pipe. In order to preclude the additional dose contribution from embedded pipe, RSNGS plans to grout piping that is greater than 2.5 inches in diameter and has residual contamination above the adjusted NRC screening levels (Table 5.19 of NUREG-5512, Volume 3) of 21,000 dpm/100 cm^2. This action level limits the dose rate to the reclamation worker to 0.55 mrem/y from Co-60 and 0.79 mrem/y from Cs-137. Grouting piping that is less than 2.5 inches in diameter results in a reduction of only 1.0 percent of the annual dose limit (See DTBE-05-009 and DTBD-05-013 for additional information), Grouting pipe that exceeds the grouting level is evaluated as outlined in Section 2.5.3.1on a case by case basis depending on the level and extent of activity and the pipe diameter.

6.7 Derivation of Area Factors

As stated in NUREG-1757, Volume 2, the $DCGL_W$ is the average concentration across an area that is calculated to result in the average member of the critical group receiving a dose at the appropriate dose limit. The general assumption is that the concentration of the radionuclides in the source is fairly homogenous. The degree to which any single localized area can be elevated above the average, assuming the average is at the $DCGL_W$, and not invalidate the homogenous assumption is characterized by the $DCGL_{EMC}$. One method for determining values for the $DCGL_{EMC}$ is to modify the $DCGL_W$ using a correction factor that accounts for the difference in area and the resulting change in dose. The area factor is then the magnitude by which the concentration within the small area of elevated activity can exceed $DCGL_W$ while maintaining compliance with the release criterion.

An area factor for use in elevated measurement comparison during final status surveys is defined by Equation 6-6.

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

Equation 6-6

where:

$DCGL_W$ = Baseline average DCGL value, and
$DCGL_{EMC}$ = Elevated measurement comparison DCGL value

NUREG-1505, "A Nonparametric Statistical Methodology for the Design and Analysis of Final Status Decommissioning Surveys," [Reference 6-29] provides the methodology for calculating area factors in Chapter 8. Chapter 8 states that the area factors should be calculated using dose pathway models and assumptions that are consistent with those used to calculate the $DCGL_W$. Area factors are computed by taking the ratio of the dose per unit concentration calculated by RESRAD or RESRAD-BUILD for the baseline area to that calculated for various smaller areas.

6.7.1 Area Factors for Rancho Seco Surface Soils

6.7.1.1 Radionuclides of Concern for Surface Soils

A site-specific suite of potential radionuclides for use at Rancho Seco was derived in Section 6.3. Of the suite of 26 potential radionuclides, only six radionuclides were positively identified. These were C-14, Co-60, Ni-63, Sr-90, Cs-134, and Cs-137. Single nuclide DCGL concentration values (each radionuclide DCGL concentration represents 25 millirem per year) were derived for a baseline default area of 10,000 m^2 in Section 6.6.2 for each of the six detected radionuclides. These single nuclide DCGL concentration values are provided in

Table 6-14 below. Area factors are calculated in this section only for the six radionuclides for which Section 6.6.2 derived DCGLs.

Table 6-14

Single Nuclide DCGL Values for Detectable Radionuclides

Radionuclide	Peak of the Mean Dose (mrem/y per pCi/g)	DCGL (pCi/g)
C-14	2.93E-06	8.33E+06
Co-60	1.93E+00	1.26E+01
Ni-63	1.60E-06	1.52E+07
Sr-90	3.76E-03	6.49E+03
Cs-134	1.09E+00	2.24E+01
Cs-137	4.62E-01	5.28E+01

6.7.1.2 Mathematical Hydrogeological Model

The mathematical hydrogeological model developed in Section 6.6.2.1 was used to calculate area factors for surface soils.

6.7.1.3 Calculation of Dose to Source Ratios for Surface Soil Area Factors

Dose to source ratios (DSRs) for the detectable radionuclides of concern were calculated by performing individual RESRAD probabilistic calculations for each of the six detectable radionuclides for each of nine specified contaminated area sizes. The site-specific RESRAD v6.22 dose model was first configured with the simplified mathematical model parameters contained in Appendix 6-S then with the statistical parameter distributions provided in Appendix 6-T. Sensitive parameters identified in Section 6.6.2.3 (density of the contaminated zone, contaminated zone K_d value for Cs-137 and external gamma shielding factor) were treated deterministically using the sensitive parameter values listed in Appendix 6-S. Parameters that were not sensitive were treated stochastically using the statistical parameter distributions contained in Appendix 6-T. RESRAD was then run in the probabilistic mode for each detected radionuclide and for each of the nine specified contaminated area sizes. A new value for the parameter "length of contaminated zone parallel to the aquifer flow" was used each time the contaminated area size was changed. The uncertainty analysis input settings for these calculations were:

- Latin Hypercube sampling,
- Random seed – 1000,
- Number of observations – 300,
- Number of repetitions – 1, and
- Grouping of observations – correlated or uncorrelated.

These calculations provided the peak of the mean DSR in mrem/year per pCi/g for each detected radionuclide. These DSRs are listed in Table 6-15.

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Incremental project and the need to mitigate additional increases to future Annual Trust Fund contributions, District staff put together a plan for continuing decommissioning through license termination, with the goal to complete decommissioning in 2008. The Board approved this plan in July 1999, and the District shifted from Incremental Decommissioning to Decommissioning.

Fuel movement began in May of 2001and by August 2002 all of the spent fuel was stored in the ISFSI under a separate Part 72 license followed by the Greater Than Class C (GTCC) generated during the Reactor Vessel Internals phase of decommissioning is 2006. In 2006 SMUD also submitted the License Termination Plan for the Rancho Seco Nuclear Generating Station (License DPR-54) which was subsequently approved by the NRC in November of 2007. In December 2008, SMUD completed all Phase I decommissioning activities and following the completion of the Final Status Surveys, submitted a letter to the NRC requesting release of the phase I portion of the site in June of 2009. On September 25, 2009, the NRC approved release of the land as requested. As of that date, the land licensed under 10 CFR 50 is an approximately 1-acre fenced parcel containing the Interim Onsite Storage Building that houses the stored low-level radioactive waste.

In 2013 SMUD determined Waste Control Specialists to be an acceptable disposal site for the class B & C radioactive waste remaining in storage within the IOSB. These wastes are currently being shipped for disposal, with completion scheduled for the fourth quarter of 2014. Once the class B & C wastes are removed from the IOSB, phase II of the decommissioning (i.e. IOSB) will commence followed by Phase II Final Status Surveys of the IOSB and its licensed footprint. After completion of FSS, SMUD will submit the request to release the remainder of the site and terminate the 10 CFR Part 50 license. These activities are schedule to be completed in 2016.

7.2 Decommissioning Cost Estimate

7.2.1 Cost Estimate Description & Methodology

The decommissioning cost estimate is prepared to satisfy the requirements of Title 10 of the Code of Federal Regulations, Part 50.75. The origin of this cost estimate is the Area-Based Decommissioning Cost Estimate prepared in 1999 and later updated in the year 2000 by TLG. Subsequently, District staff updated the estimate in the year 2001, 2002, 2003, 2004 and again in the year 2005 [Reference 7-3]. Each of these updates prepared by District staff was reviewed by TLG and, as such, is utilizing the current 2005 estimate updated with actual cost and forecast data as the basis for the cost estimate in this submittal of the LTP.

The methodology used to develop the cost estimate follows the basic approach originally presented in the Atomic Industrial Forum (now Nuclear Energy Institute) program for developing standardized decommissioning cost estimates published as AIF/NESP-036, "Guidelines for Producing Commercial Nuclear Power Plant Decommissioning Cost Estimates," [Reference 7-4]. This document presents a unit cost factor method for estimating direct activity costs, activity by activity, simplifying the estimating process. Unit factors for the removal of equipment, concrete, steel, etc., were constructed from site-specific labor costs provided by the District. The unit factors are based upon labor costs currently being used as part of the incremental decommissioning project. The direct activity costs were then estimated using the plant inventory developed for each work area.

The unit cost factor method provides a demonstrable basis for establishing reliable cost estimates. The detail available in the unit cost factors for activity time, labor costs (by craft), and equipment and consumable costs provides assurance that cost elements have not been omitted. The detailed unit cost factor, coupled with the plant-specific inventory of piping, components, and structures, provide a high degree of confidence in the reliability of the cost estimate.

To account for the unique working conditions associated with decommissioning, work difficulty factors (WDFs) were assigned to each work area. WDFs are commensurate with the inefficiencies associated with working in confined, hazardous environments and are applied as increases to the unit cost factors. The WDFs take into account factors associated with access difficulties, use of respiratory protection, Radiation Protection/ALARA, use of protective clothing and accounting for work breaks. These factors and their associated range of values were developed in conjunction with the Atomic Industrial Forum's Guidelines Study.

The decommissioning plan schedule was used to determine the period-dependent costs for program management, administration, field engineering, equipment rental, contracted services, etc. The study relies upon site-specific salary and wage rates for the personnel associated with the intended program.

TLG's cost model is comprised of a multitude of distinct cost line items, calculated using cost factor methodology described earlier. Period-dependent and collateral costs are combined to produce a comprehensive accounting of the identified expenditures. However, the resulting costs in and of themselves do not comprise the total cost to accomplish the project goal of license termination.

Consistent with industry practice, contingencies were applied to the decontamination and dismantlement costs developed as specific provision for unforeseeable elements of cost within the defined project scope, particularly important where previous experience has shown that unforeseeable events that will increase costs are likely to occur. The cost elements in the estimate are based on ideal conditions; therefore, the types of unforeseeable events that are almost certain to occur in decommissioning, based on industry experience, are addressed through a percentage contingency applied on a line item basis. The contingency, as used in the estimate, does not account for price escalation and inflation in the cost of decommissioning over the remaining project duration.

7.2.2 Summary of the Site Specific Decommissioning Cost Estimate

The decommissioning cost estimate in total is defined as the funding required to complete decommissioning, however, the cost assigned to a given line item within the estimate is not as rigorously defended. A basic assumption of the estimating process is that when specific line items have been over-estimated, the unspent funds will be required to cover the costs associated with other line items that have been under-estimated. Historically, the overall impact is that the cost of work completed to date has been, in general, over-estimated. This has resulted in funds that were not required to offset the actual costs incurred in completing work. However, the presupposition of the correctness of the total estimated cost requires that these funds be preserved for future work. The remaining cost projected to complete the decommissioning of Rancho Seco is $138.3[2] million for the period 2006 through Phase I site release in 2008, with additional amounts of $24.7 million for the transfer of GTCC waste to the DOE in 2027, oversight of waste stored in the Interim Onsite Storage Building (IOSB) through 2028, and Phase II license termination in 2028. The total cost for decommissioning, including previously

[2] From the current Cost Estimate, Reference 7-3

expended funds, is $534 million (to-date actual costs in the year spent dollars plus future work in year 2005 dollars). A summary of the remaining major cost contributors is provided in Table 7-1 and graphically in Figure 7-1.

The cost estimate provides an overall cost for the duration of the project including all costs incurred after transitioning from operating and maintenance (O&M)-financed expenses after plant shutdown through 10 CFR 50 license termination, plus an amount to cover District costs anticipated for transferring control of the used nuclear fuel to the DOE. The costs contained in this cost estimate can be generally grouped into four basic categories. These are: technical decommissioning costs; non-technical District costs; the staffing plan; and fuel dry storage project costs.

The section of the cost estimate based upon detailed engineering calculations is the technical portion of the decommissioning cost estimate. This portion is based upon engineering calculations that use a variety of input factors, which include the following:

- Unit cost factors for removal;
- Inventories of plant systems and components remaining after the Incremental Decommissioning project;
- Difficulty factors involving the level of effort required and the ability to physically access the material;
- Impacts due to radiological conditions (both radiation and contamination); and
- The presence of hazardous materials (e.g., lead-based coatings, asbestos insulation).

The technical costs include the direct costs of dismantlement and the indirect costs including generation of incidental radioactive waste, required health physics supplies, small tool allowances, and other costs in the "Undistributed" category. The basis for the technical decommissioning costs remains the 2000 Cost Estimate Update prepared by TLG, except when specific costs are updated based upon additional data such as recent industry or site experience.

The Area Based Decommissioning Cost Estimate prepared by TLG Services in 1999 and subsequently updated in 2000 is the basis for the LTP cost estimate for Rancho Seco. The estimated total cost is $534 million which is the sum of previously expended funds in the dollars for the year spent, plus future costs in 2005 dollars. For budgetary and financial planning purposes, this estimate has escalated annually for inflation at a average rate of 2.7% for general costs and 3% for staffing costs.

Technical costs are now updated using the basic methodology described above. The basis for the technical costs remains that used for the 1999 Area-Based Cost Estimate with long-term contract information as provided in the 2000 update. Both the 1999 Estimate and 2000 Update were performed by TLG.

In certain instances, line item values have been changed to reflect an increased level of detail in work planning. The changes are made by redistributing available funds among a larger number of detailed line items, however, the total costs remain consistent with previous estimates and the update methodology described. In these cases, the changes reflect the increased level of detail in the scheduling software and maintain consistency between the scheduling software and the cost estimate.

Non-technical District costs are those associated with facility maintenance, District overhead, travel to professional seminars, and other costs not directly derived from the decommissioning process. These costs are determined through the annual budgeting process, and are forecast through the end of the project based upon historical data. The schedule of the technical portion of the project provides the basis for determining the non-technical costs.

A major contributor to the overall cost of decommissioning is the staff cost. The cost of staff is based upon the staffing plan developed to meet the decommissioning schedule and needs of the project in terms of staffing levels, and also based upon the actual and projected staffing costs derived from current contracts and the budgeting process. Also included are additional staff costs required to oversee the radioactive waste stored in the IOSB until shipped for disposal.

Fuel dry storage project costs include fuel storage costs through 2008 and the cost of transferring the GTCC material, which will be stored until transfer with the fuel in the ISFSI, to the DOE. The transfer of the GTCC material is tied to the fuel storage because it is assumed the GTCC material would be placed into the same repository as the fuel when the DOE develops the repository.

Consistent with the NRC definition of decommissioning under 10 CFR 50.2, the radiological decommissioning costs consider those costs that are associated with normal decommissioning activities necessary for termination of the Part 50 license and release of the site for unrestricted use. Additionally, the Cost Estimate includes costs for fuel storage through 2008, coinciding with the scheduled completion of phase one of License Termination. The Cost Estimate does not include costs associated with the disposal of non-radiological materials or structures beyond that necessary to terminate the Part 50 license.

Table 7-1

Summary of Remaining Decommissioning Costs
In Year 2005 Dollars (thousands of dollars)

Work Category	Cost in 2005$ (2006 & beyond)	Remaining Costs
Decontamination	2,663	1.6%
Large Components, RB Concrete	28,429	17.4%
Transportation	2,768	1.7%
Waste Disposal	7,126	4.4%
Characterization/Remediation	14,961	9.2%
Final Status Survey	13,434	8.2%
Project Staffing	52,730	32.3%
Materials and Equipment	3,278	2.0%
Insurance	1,156	0.7%
Other Undistributed Costs	12,811	7.9%
Contract & Material Surcharges	823	0.5%
Stored Waste Oversight	1,994	1.2%
Class B, C, & GTCC Disposal Costs	20,552	12.6%
Total	163,088	100.0%
Expended thru 2005	371,097	
Grand Total	534,185	

7.3 Decommissioning Funding Plan

The District had maintained an internal decommissioning fund since the early 1980's. In 1991, the District transferred $90 million from the internal fund into an "external sinking decommissioning trust fund" and submitted its Financial Assurance Plan to the NRC describing the use of the external sinking fund. There have been no significant modifications to the external sinking fund since the initial submittal.

The District plans to accumulate funds in the external trust fund, at the rate of $27 million per year, through 2008. Based on the current decommissioning cost estimate and funding rate, collecting funds through 2008 will provide sufficient funds to complete decommissioning Rancho Seco and terminate the 10 CFR Part 50 license.

The external trust fund is currently maintained by Wells Fargo Bank. The balance is reviewed on an annual basis to ensure the adequacy of the annual contribution to assure funds will be available to complete decommissioning and terminate the 10CFR Part 50 license.

The District has concluded that the current estimate forecast is adequate to complete the remaining decommissioning activities for Rancho Seco. Actual costs are monitored continuously against estimated costs. The Cost Estimate is updated annually per 10 CFR 50.75(b)(2) and reflects impacts such as industry experience and items identified by the monitoring process.

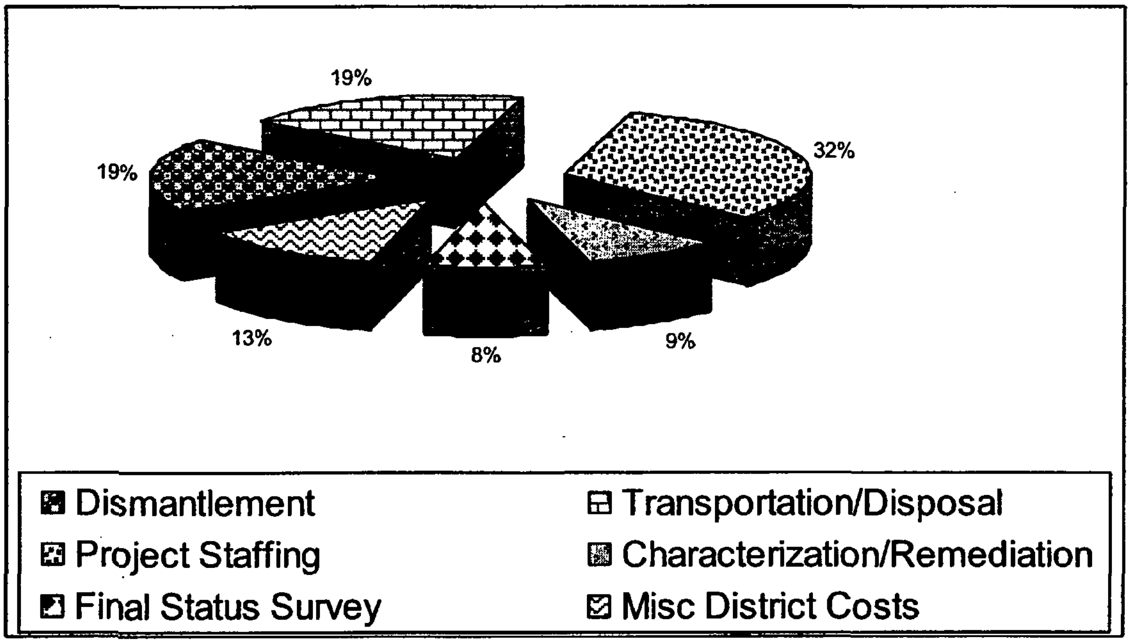

Figure 7-1
Summary of Remaining Decommissioning Costs in Year 2005 Dollars