ELECTRIC POWER RESEARCH INSTITUTE

Rancho Seco Nuclear Generating Station Decommissioning Experience Report

Detailed Experiences 1989-2007

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Final Report, December 2007

EPRI Project Manager
K. Kim

ELECTRIC POWER RESEARCH INSTITUTE
3420 Hillview Avenue, Palo Alto, California 94304-1338 • PO Box 10412, Palo Alto, California 94303-0813 • USA
800.313.3774 • 650.855.2121 • askepri@epri.com • www.epri.com

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ORGANIZATION(S) THAT PREPARED THIS DOCUMENT

NewRad, Inc.

Suncoast Solutions, Inc.

NOTE

For further information about EPRI, call the EPRI Customer Assistance Center at 800.313.3774 or e-mail askepri@epri.com.

CITATIONS

This report was prepared by

NewRad, Inc.
21452 Robin Ln.
Pine Grove, CA 95665

Principal Investigator
J. Newey

Suncoast Solutions, LLC
1348 New Forest Lane
Osprey, FL 34229

Principal Investigator
J. Newey

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REPORT SUMMARY

Several U.S. nuclear power plants entered decommissioning in the 1990s. Based on current information, the next group of plants whose license will expire will not begin decommissioning for nearly a decade. This report provides detailed information on the decommissioning of one plant, the Rancho Seco Nuclear Generating Station, in order to capture its experience for future plants.

Background

Rancho Seco is a 913-megawatt Babcock & Wilcox design nuclear power plant owned by the Sacramento Municipal Utility District that began commercial operation in 1975. It was shut down in June 1989 as the result of a voter referendum. Due to a minimal decommissioning fund balance, the decision was made to enter an extended period of SAFSTOR to allow the activity to decay and the decommissioning fund to accrue to a level that is required for dismantlement. The projected start for decommissioning is 2008.

Objectives

- To summarize the decommissioning experience of Rancho Seco Nuclear Generating Station
- To provide lessons learned for future plants entering decommissioning.

Approach

The project team gathered survey information from managers at current decommissioning facilities to determine areas of interest to future decommissioning managers. With the approval of Rancho Seco management, much of the information contained in this report was retrieved from the personal files of the principal investigator, who was also a contract employee at Rancho Seco during the time of the preparation of the report. The project team gathered additional information from onsite interviews with several Rancho Seco managers and through other references and sources. The focus was on defining specific lessons learned for use at future plants entering decommissioning and making recommendations for current operating plants to improve performance for future decommissioning.

Results

A similar effort was performed by EPRI to report decommissioning experiences at the Connecticut Yankee (EPRI Report 1013511) and Maine Yankee (EPRI Report 1011734) plants. The report presents the decommissioning experience and lessons learned at Rancho Seco in the following areas:

- Pre-shutdown actions and analyses
- Transition activities from operations to decommissioning
- Use of Decommissioning Sub-Contractors
- Fuel Storage Options
- Regulatory and Stakeholder interaction
- Specific Technologies Used
- Site Closure Issues

EPRI Perspective

One of the key objectives of the EPRI Decommissioning Technology Program is to capture the good practices and lessons learned from the plants currently being decommissioned. Because several major plant-decommissioning programs are nearing successful conclusions, EPRI is documenting relevant experiences to aid future decommissioning activities, both in the United States and internationally.

Keywords

Decommissioning
Site Closure
License Termination

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- Einar Ronningen – Dismantlement Superintendent – Radiological
- Wayne Hawley – Dismantlement Superintendent – Operations
- Robert Jones – Nuclear Regulatory Coordinating Superintendent
- Michael Snyder – Principal Decommissioning Radiological Engineer

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1 INTRODUCTION

1.1 Background

Over the past ten years, EPRI has developed and published a number of experience reports and workshop proceedings related to decommissioning. These documents were developed in conjunction with U.S. nuclear plants over a period of nine years. Many of the reports reflected work performed at various times and phases of decommissioning. The EPRI decommissioning library provides a wealth of knowledge related to decommissioning tasks and regulatory requirements. These documents and proceedings will provide a sound reference base for future nuclear reactor facilities as undergo decommissioning.

As of 2007, the majority of these U.S. nuclear reactor facilities (except for those that have chosen to place their plants into SAFSTOR and decommission at a future date) have completed most if not all of the required decontamination and remediation and anticipate the full conclusion of their decommissioning projects in the near term. Based on currently announced or submitted license extension applications, only a few additional U.S. reactors will enter decommissioning prior to 2020.

EPRI undertook the publishing of Plant Experience Reports to provide an overview of the activities and experiences in the decommissioning of specific plants. These reports provide a summary of plant decommissioning information for use in future decommissioning projects, Rancho Seco Experience report will be the third in this series. Similar reports have been published for Maine Yankee (EPRI Report 1011734) and Connecticut Yankee (EPRI Report 1013511.)

The “essential information” for these reports includes such topics as;

- detailed project plans
- schedules
- engineering analysis
- decommissioning costs and personnel exposure
- and applied technologies and experiences for decommissioning activities.

A second level of information is included on so-called “soft areas” including:

- stakeholder interaction
- regulatory interaction

- and project decision methods (e.g., use of decommissioning operations contractor or not, wet or dry spent fuel storage, or decommissioning approach).

Maine Yankee Atomic Power Company hosted the pilot detailed experience report. The Maine Yankee report has been completed and is listed as Reference [1][1]. A similar report was also completed for the Connecticut Yankee Plant decommissioning and is listed as Reference [2]. The Connecticut Yankee report paralleled the Maine Yankee work and provided a guide for the Rancho Seco Plant decommissioning experience report. In order to allow ease in comparing the experiences of the three plants, the format and sequence of topics are generally maintained in the following decommissioning experience report for Rancho Seco Nuclear Generating Station (RSNGS).

Information relative to the RSNGS decommissioning was supplemented by site interviews which were conducted at the Rancho Seco site in 2006 and 2007. Interviewees included the Decommissioning Demolition Manager, the Decommissioning Technical Support Manager the Licensing Manager and the Radioactive Waste Supervisor.

The remainder of this document provides a brief summary of the RSNGS decommissioning project followed by summaries for each of the following topics:

- Pre-Shutdown Issues
- Transition Activities
- Decommissioning Subcontractors
- Fuel Storage Options
- Water Management
- Regulator and Stakeholder Interaction
- Engineering and Use of Technology
- Long Term Site Issues
- Groundwater Contamination Investigations
- LTP and Final Survey Process

Chapter 10 summarizes the decommissioning lessons learned from Rancho Seco. In addition, data related to the decommissioning project is provided in various appendices:

- A summary of radioactive and non-radioactive waste expected to have been shipped through the end of the project is provided in Appendix A.
- A cost overview is provided in Appendix B.
- A project timeline is provided in Appendix C.
- A summary of radiation exposure is provided in Appendix D.

[1] The list of references can be found in Chapter 11.

1.2 Rancho Seco Overview

Rancho Seco is a 913-megawatt Babcock & Wilcox (B&W) design nuclear power plant owned by the Sacramento Municipal Utility District (the District) that began commercial operation in 1975. It was shut down in June 1989 as the result of a voter referendum. Due to a minimal decommissioning fund balance, the decision was made to enter an extended period of SAFSTOR to allow the activity to decay and the decommissioning fund to accrue to a level that is required for dismantlement. The initial projected start for decommissioning was 2008 which coincided with the duration of the operating license.

In 1991, the decision was made to place the spent fuel into dry storage and drain and de-energize systems to enter an extended SAFSTOR condition. This change would allow a major reduction in the required staff. An Independent Spent Fuel Storage Installation (ISFSI) was built and contracts for casks and fuel storage liners were put in place. However numerous delays concerning the design and fabrication of the components continued to postpone fuel transfer. Fuel transfer was finally completed in August 2002 with 21 fuel canisters being placed in the ISFSI.

With the operating staff facing an extended waiting period for fuel movement to the ISFSI, there was a possibility for significant cost savings by using plant personnel to begin removal of low activity components for disposal at the Envirocare[2] of Utah disposal site. This approach was proposed as a three-year “incremental decommissioning project” focused on the dismantlement of the Turbine Building systems and a portion of the Tank Farm systems. The project was approved for a 1997 start, with annual renewals based on performance. This approach proved to be highly successfully and lead to approval of full dismantlement in July 1999 with a completion date of 2008 which corresponded to the final payment to the decommissioning fund.

The dismantlement will leave the concrete structures (reactor building and auxiliary building) as permanently sealed empty shells. The Interim Onsite Storage Building (IOSB) will remain containing the Class B and C waste awaiting final disposal. Following the termination of the plant operating license a Part 50 license will only continue for the life of the IOSB building, beginning in 2008. Once an acceptable waste site is available for the Class B and C waste, the waste will be shipped and the remaining licensed area will be surveyed and released.

The site is approximately 25 miles (40 km) southeast of Sacramento and 26 miles (42 km) northeast of Stockton in the central valley of California between the foothills of the Sierra Nevada Mountains to the east and the Pacific Coast range bordering the Pacific Ocean to the west. A map of the facility and location is included as Figure 1-1. The RSNGS site consists of an approximately 87-acre fence-enclosed Industrial Area containing the nuclear facility surrounded by District owned and controlled property with a total area of 2,480 acres.

The Sacramento Municipal District will continue to use the site for power producing activities. The office buildings will be occupied and the switchyard will remain in-service. The District constructed a 30-acre natural gas-fired power plant on the RSNGS site, approximately ½ mile

[2] As of 2006, Envirocare Care of Utah is owned by Energy*Solutions*. However, in this report, the facility will be referred to as Envirocare or Envirocare of Utah as this was the title of the facility during the RSNGS decommissioning project activities.

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; and the 10 acre 10 CFR Part 72 licensed Independent Spent Fuel Storage Installation (ISFSI).

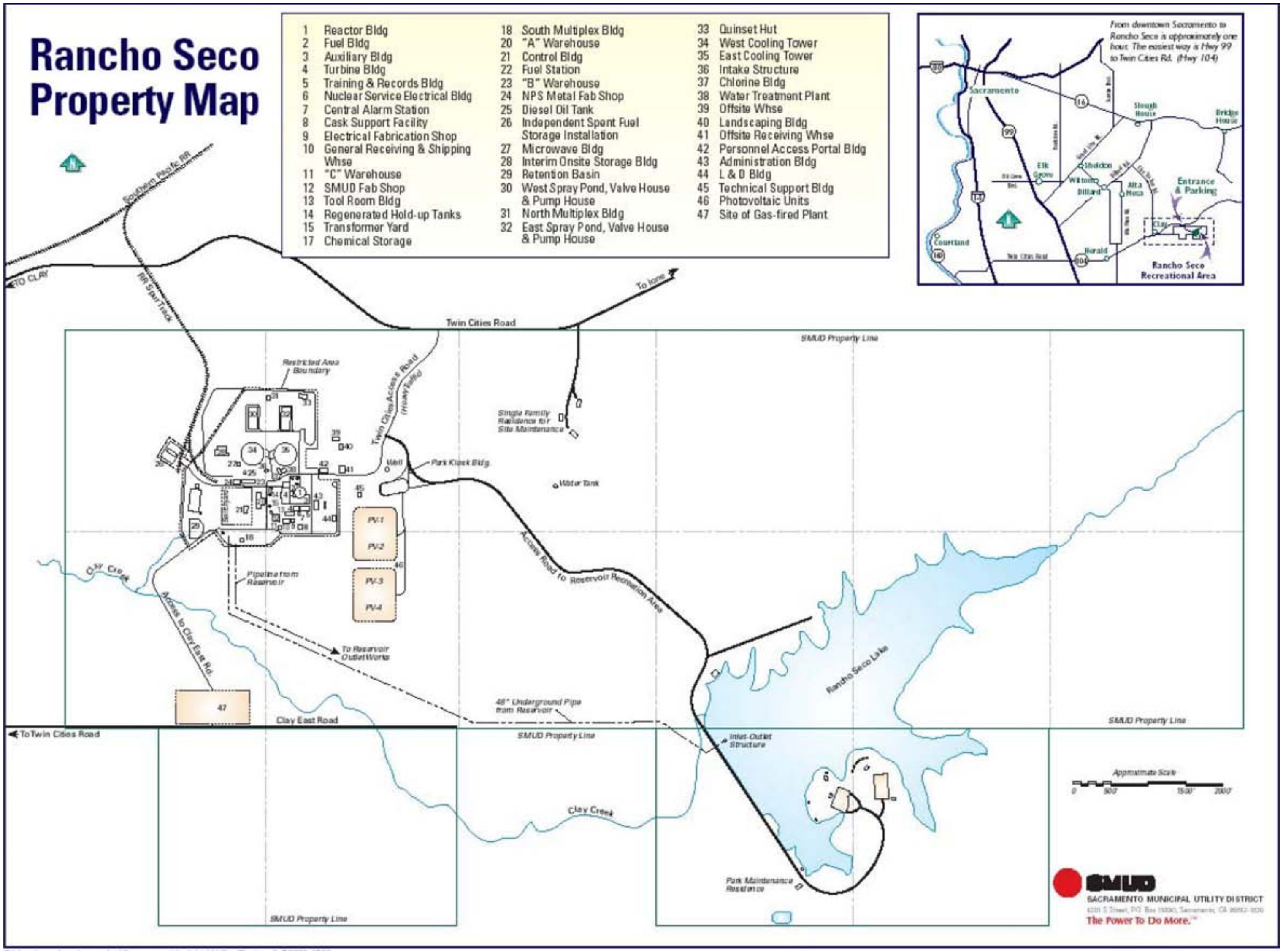

Figure 1-1
Rancho Seco Nuclear Generating Station Site Map

2 PRE-SHUTDOWN ISSUES

2.1 Shutdown Decision

The Sacramento Municipal Utility District, Rancho Seco Nuclear Generating Station (RSNGS) went into commercial operation on March 1, 1975. The plant operated somewhat erratically through 1985. At that time a shutdown incident precipitated an NRC mandated re-start program. The re-start program involved significant effort and upgrade cost. The plant was re-started in 1988. During that period the plant achieved an overall capacity factor of approximately 50 percent. However, due to the public nature of the utility the District called for a referendum on the continued operation of the Rancho Seco plant. This referendum by voters failed and the required plant shutdown took place on June 7, 1989.

After shutdown the Rancho Seco plant was initially maintained for a possible buyer. Soon it became apparent that no buyer would appear due to the requirement for another voter referendum to allow restart. The District moved to permanent plant shutdown. Reactor defueling was completed on December 8, 1989 with the required decommissioning license documents were approved for the permanently shut-down and defueled reactor in 1990.

2.2 Pre-Shutdown Planning

The permanent shutdown of RSNGS in 1989 was unplanned. In fact new fuel had been ordered and a new main turbine was on-site ready to be installed at the next refueling outage. The spent fuel pool had been re-racked in 1983 increasing the capacity to allow operation to the then planned permanent shutdown in 2008.

At the time of permanent shutdown the only major efforts for decommissioning that had been conducted included the following:

- The Decommissioning Fund was minimally funded.
- As required by U.S. Nuclear Regulatory Commission (NRC), a Title 10 Code of Federal Regulations Part 50.75 (g) listing of plant occurrences such as spills and other contamination incidents had been established.

As will be discussed in the next section, this level of preparations for a decommissioning was not adequate to prepare the plant for a permanent shutdown. For example:

- A decommissioning cost estimate had not been prepared. Once one was prepared it utilized standardized unit cost factors based on a standard plant design. The approach reflected in the cost estimate was a high level estimate and did not in all cases reflect the structures and systems actually present at RSNGS. A more detailed plan would have better prepared RSNGS for an unexpected permanent shutdown.

- The 10 CFR 50.75 (g) file that had been maintained did not capture all of the events important to a complete and meaningful Historical Site Assessment. As will be discussed later, an extensive effort was needed in the later stages of the RSNGS decommissioning to prepare an adequate Historical Site Assessment (HSA). If the 50.75 (g) file had been more completely prepared during the operation of the plant, it would have been a significant benefit in early decommissioning planning.

3 TRANSITION ACTIVITIES

3.1 Overview

The period after the announcement that the nuclear power plant would be permanently shutdown was a very busy one. In the case of Rancho Seco, where the shutdown was unplanned, this transition period was even more demanding. Some of the key tasks that are normally accomplished during this period are:

- Determining the decommissioning option (SAFSTOR, DECON, etc,)
- Managing human resources for operating plant personnel and developing the transition and ultimately the decommissioning organization
- Preparing the initial licensing documents such as the Post Shutdown Decommissioning Activities Report (PSDAR), or in the case of Rancho Seco a Decommissioning Plan, and conducting a public meeting
- Preparing a Historical Site Assessment for the facility (not required at the time of RSNGS shutdown)

The remainder of this section will address the experiences at RSNGS that occurred during the period directly following the shutdown decision.

3.1.1 Transition Period Staffing Issues

One of the first tasks to be faced once the permanent shutdown of a plant is scheduled is to address human resources issues. With the shutdown being unplanned at RSNGS, the question in many workers mind was: “What about my job?” The human resource issues related to the shutdown of a nuclear plant needed to be addressed to help avoid personnel errors and injuries that could result from distracted workers. There also needed to be a plan to retain key employees that have the institutional plant knowledge that would be needed during the decommissioning.

Within weeks after the initial shutdown non-essential contractors were terminated. Fairly quickly a voluntary reduction program began for SMUD non-essential personnel. However, much of the operating staff was retained through the transition period awaiting the fuel transfer operations. As the fuel transfer to dry storage was delayed, significant resources were being spent on maintaining the idle staff. In order to best utilize the staff during this transition period, incremental decommissioning was instituted. Incremental decommissioning and its impact on cost savings is discussed in section 3.4 below.

3.1.2 Transition Licensing Actions

The first licensing activity after the decision to permanently shutdown is to notify the NRC of RSNGS's intent to permanently cease operations and that the reactor was completely defueled. These certifications were made to the NRC on December 5, 1989.

3.1.3 Decommissioning Plan

A Decommissioning Plan is a summary document that outlines the overall plan that a licensee will use to carry out the decommissioning of the shutdown plant. The RSNGS Decommissioning Plan was issued by SMUD on August 22, 1992**.** The Plan for Rancho Seco contained the following information (a general description on the section content is presented under each section title)

- Description of Planned Decommissioning Activities
- Major Decommissioning Activities
- General Description of Decommissioning Methods
- Estimated Personnel Radiation Exposure
- Estimated Radioactive Waste Quantities to be Generated
- Decommissioning Schedule
- Estimated Decommissioning Costs
- Environmental Impacts

As can be seen from the listing of information provided above, sufficient planning and estimating needed to be performed to determine the information to be contained in the Plan. In order to accomplish this, a team of specialists was formed to prepare this document. This task included the following:

- Review existing plant programs to assess their applicability to decommissioning
- Review and reclassify systems important to decommissioning operations
- Revise procedures and license basis documents to reflect the plant's defueled and permanently shutdown configuration
- Determine size and quantities of plant systems and structures to allow waste volume calculations
- Conduct scoping characterization surveys to determine the radiological status of plant systems and structures and the volume of contaminated soil on site

The preparation of the above information was a substantial task and was performed during the eight month period between the decision to decommission and the issuance of the Decommissioning Plan.

3.1.4 Public Meeting

After the issuance of a Decommissioning Plan, the NRC conducts a public meeting in the vicinity of a reactor, normally within 90 days of the receipt of the document. The meeting provides a forum for the conveyance of information such as that contained in the Decommissioning Plan to the public. The NRC also provides a discussion of the regulatory and oversight approach to be utilized during the decommissioning. Two meetings were conducted in the town of Galt, California and at SMUD headquarters in Sacramento in the fall of 1991.

3.2 Early Decommissioning Projects

3.2.1 Site Characterization

An initial site characterization was performed in support of the Decommissioning Plan. This characterization provided the basis for initial planning for incremental decommissioning. An activation analysis was also performed to estimate curie content of major reactor systems.

3.2.2 Asbestos Removal

The removal of asbestos insulation is a project that can be started very early in the decommissioning. Rancho Seco found this beneficial for the following reasons:

- The removal of asbestos in most areas generally required little regulatory input or Health Physics support.
- It could be accomplished in many separate pieces – starting and stopping as cost and schedule dictate.
- The removal of asbestos generally requires the isolation of the work area.
- The removal of asbestos often cannot be carried out at the same time as the removal of the associated systems or piping. Removal of the asbestos insulation would clear the way for removal of systems at a later date.

Rancho Seco began asbestos removal activities soon after the permanent decommissioning decision. The largest portion of the major asbestos removal activities involved the removal of asbestos from the cooling towers, this was accomplished in 1995.

3.2.3 Hazardous Material Removal and Remediation

Early in the SAFSTOR period a hazardous material removal and remediation program was performed. In addition to the asbestos removal all hazardous materials were catalogued and, if necessary, removed. Underground tanks were removed and sampling and remediation was performed in non-radiological areas.

3.2.4 Reactor Coolant System Chemical Decontamination

No system decontamination was performed at Rancho Seco for the following of reasons:

- At the time the shutdown decision was made dismantlement was originally planned for 2008, therefore system decontamination provided no immediate benefit.
- Once dismantlement began in 1998 the conditions of the systems were such that significant maintenance would have been required prior to the decontamination process.
- Additionally, after 10 years of decay there was little cost benefit in the expected dose-rate reductions.

3.3 Dry Fuel Storage

In 1991, the decision was made to place the spent fuel into dry storage, allowing the plant to enter a “hardened” SAFSTOR condition and allowing significant reduction in the required staff. An ISFSI was built and contracts for casks and fuel storage liners were put in place; however numerous delays continued to postpone fuel transfer. Fuel transfer was not completed until mid 2002.

3.4 Incremental Decommissioning

The original baseline decommissioning cost estimate used a value of $405 per cubic foot for waste disposal at the *planned* Ward Valley disposal site. While this value was not valid, because the site did not open, it was used for comparison purposes. In 1995 the Envirocare disposal facility became available as an option for disposal of very low activity waste. The Envirocare waste cost was significantly below that estimated for Ward Valley. The Envirocare facility provided an opportunity for significant savings for disposal of very low activity waste, such as steam and cooling systems in the Turbine Building, which had become contaminated from minor system-to-system leaks. Studies also showed that a significant portion of the waste in the Auxiliary Building and the Reactor Building would qualify for disposal at Envirocare. The Turbine Building was selected for initial dismantlement activities based on the large volumes of potentially contaminated materials with expected very low activity levels. This work could be performed with minimal radiological controls and oversight.

With the staff waiting for fuel movement and the possibility for significant cost savings, a three-year incremental decommissioning project was proposed to dismantle the Turbine Building systems and a portion of the Tank Farm systems. The project was approved to begin in 1997 with annual renewals based on performance.

Before actual dismantlement could begin significant up-front work was required. Contracts were needed for disposal, shipping, waste processing and labor. Engineering was required for abandonment of systems and components and the necessary isolations from active systems. Procedures governing dismantlement were also required.

An interdisciplinary team of loaned employees was formed to manage the work and the waste. The team included personnel from the radiation protection, operations, maintenance and engineering groups. Specialized waste and decommissioning personnel were brought in to supplement the group. Dismantlement activities began with site personnel as soon as procedures and engineering were in place. It then took more than a year to get the required contracts and additional specialized employees in place before waste could be shipped.

Contracts were required for waste processing, waste disposal, waste shipping, contract labor and equipment, asbestos abatement, lead paint abatement, and specialized personnel. All of these contracts were competitively bid resulting in long lead times prior to commencement of work. The initial dismantlement work was performed by Rancho Seco maintenance personnel. As larger components were removed, the workforce was supplemented by a local crane and rigging company (Bragg Crane and Rigging. Later in the project J.A. Jones Construction Services was selected to provide the dismantlement personnel, material needs and deconstruction oversight. GTS/Duratek[3] was selected to provide waste services and specialized personnel. Frank W. Hake and Associates was also selected to provide waste processing services. A contract was put in place with Envirocare of Utah, Inc. for waste disposal services.

Previous characterization work had determined that most Turbine Building systems would be non-radioactive with the exception of the Turbine Plant Cooling Water System, Main Steam, Auxiliary Steam, First and Second Point Heaters, Reheaters and the Turbine. However, all systems were removed to simplify final survey activities. Updated guidance from the NRC indicated that “systems and components” were not subject to Final Survey (ie. The Licence Termination Rule that allows residual radioactivity equivalent to 25 mrem/yr above background to remain): such components must be “released” through the licensee’s release program.

Actual dismantlement began with demineralized water, condensate polishers and chemical addition systems. Next feedwater heaters and reheaters were dismantled, followed by the turbine, feed pumps and finally the condenser. Most dismantlement of minimally contaminated components was done with standard oxyacetylene torches. Dismantlement of the equipment within the Turbine Building was completed in mid 1999, while outside work continued in the Tank Farm and pipe chase areas.

3.4.1 Survey for Release

The Incremental Decommissioning Program began with a small budget ($12 million for three years) to prove to management that SMUD personnel could manage decommissioning and save money based on the original cost estimate. One of the most effective cost-saving measures early on was the program of surveying material for release as non-contaminated. Most material was expected to be free of contamination, but all system components in the Turbine Building are

[3] As of 2006, GTS/Duratek is owned by Energy*Solutions*. However, in this report, the facility will be referred to as GTS/Duratek as this was the title of the facility during the RSNGS decommissioning project activities.

required to be surveyed as they are removed and size reduced. Some system components required minimal surveys due to system history of no internal contamination. Steam systems however required extensive surveys of all surfaces to allow free release.

If surfaces existed that were not accessible, then the procedure required that the component either must be cut open to allow survey or be disposed of as contaminated. To avoid this for some components with inaccessible areas that are expected to be clean, a procedure was developed to allow an evaluation to be done of projected contamination levels for inaccessible areas. The “Inaccessible Contamination Evaluation,” (ICE for short) procedure was developed for these components. It required system knowledge, survey of accessible areas and might include a sampling of inaccessible areas by destructive means. Items released under this program include the third and fourth point heaters, most of the auxiliary boilers and the outer turbine covers. Major portions of the condenser and the fifth and sixth point heaters were also released in this manner.

The free release program has not been without its problems. Two different incidents occurred where the radiation monitors at the scrap yards rejected truckloads of Rancho Seco scrap metal. Only one incident resulted in rejection by the recycler. In the second incident, the procedure in use at the recycler resulted in their acceptance of the material, however, through a courtesy notification, Rancho Seco management requested return of the material. (The recycler’s procedure has the truck move through the radiation monitors up to three times: if an alarm, pass through again, if no alarm the 2nd time, pass through a 3rd time. If in 3 monitoring attempts, only the 1st attempt alarmed, the facility will accept the material. Because of Rancho Seco's relationship with the scrap yard, the personnel notified Rancho Seco Management, who decided to have the material returned for further investigations.) In one case a very small area was apparently missed in the survey process. In the other, re-survey of the material by special means showed that a very small amount of activity (below levels that could be detected by normal survey procedures) was distributed over many pieces causing the scrap yard monitors to alarm. Rancho Seco had developed a release program based on Reg Guide 1.86 and subsequent guidance (Reg Guide 85-92 and Reg Guide 81-07). At the time of the incidents, the release criteria was essentially "<100 cpm above background with a frisker." After the 2nd incident, an improved interpretation of NRC guidance was implemented, and the release criteria essentially became “no detectable activity with a frisker, monitoring to be performed in areas with backgrounds <100 cpm”. The activity levels in these incidents were not high enough to be of concern over unmonitored exposure to members of the public in either case.

Survey procedures were revised to institute corrective actions that were identified. An example of a revised procedure is that each component and structure was marked with grids. Each square of the grid was surveyed and initialed to indicate completion of each grid survey. When these revised procedures were released an extensive training of personnel was required. Also, a truck monitor was purchased for site use to mimic the monitoring being conducted at the scrap yards. The truck monitor provided a final check before release to ensure that aggregate quantities could be evaluated. It should be noted that the aggregate quantity surveys were already in place, and were conducted under IAW NRC guidance. The truck monitor was a supplement to the “aggregate quantity” surveys that were in place before the incidents occurred.

These measures resulted in increased cost due to additional survey and decontamination time. However the free release of material was still the most cost-effective disposition for such material.

The survey program was responsible for the recycle of approximately 3 million pounds (1.4 million kg) of material that might have been sent for radioactive waste disposal.

3.4.2 Decontamination

A booth was installed to allow grit blasting of lightly contaminated materials. It was found to be very cost effective for high-density carbon-steel materials and was used on a large portion of the steam system components and piping that were found to be contaminated. Components were pre-sized to fit in the booth with all necessary surfaces exposed. Once the contaminated surfaces were blasted, a complete survey was performed prior to material release. Since the more contaminated materials were sent directly to packaging for disposal, few items failed the survey after blasting. Approximately 1 million pounds (453,000 kg) of material has been successfully decontaminated and sent for recycling in this manner.

Some of the components that were mostly clean, but could not easily be decontaminated or surveyed were sent offsite for processing if it was deemed economical to do so. These components included portions of the Moisture Separator Reheater (MSRs) and the first and second point heater tubes.

3.4.3 Packaging and Disposal

For those items that could not be free released, decontaminated on site or economically sent for processing, disposal was the remaining option. Disposal cost is mostly a function of volume and priced as dollars per cubic foot. Therefore, packaging efficiency or density of the burial package was the most important factor reducing the disposal cost. Standard disposal to qualify as debris requires waste to have one dimension that is not greater than 10 inches (25.4 cm.) This requires most waste material to be cut to meet this criterion. Packaging efficiency was achieved mostly by moving the material that did not pass release criteria to a staging shop where material was torch cut to a size and shape that could easily be packed. Other innovative approaches were developed, such as placing smaller pipes in larger pipes. The result was extremely heavy containers, with one of the 100 cubic foot (2.8 cubic meter) boxes exceeding 30,000 pounds (13,608 kg).

Some major components of value that were contaminated were transferred to other licensees, thus avoiding disposal cost. Included in this category were the high-pressure turbine rotor and two MSR tube bundles that were sent to another nuclear power plant. Many of the pumps and motors from the Auxiliary Building were sent to a vendor for refurbishing and sale to other plants.

3.5 Board Approval of Full Decommissioning

Based on the success of Incremental Decommissioning the SMUD board approved full decommissioning in July 1999. The scheduled completion was set as the end of 2008 to correspond to the last year of contribution to the SMUD decommissioning fund. This date was chosen to coincide with the expiration date of the Part 50 license. There was a high sensitivity to any increases in the overall cost estimate due to the dry fuel storage project having exceeded its budget. It became very important to meet budget and schedule. The unwritten agreement was that the project could be shutdown at any time with a return to SAFSTOR if problems arose.

Once the approval occurred many programs were accelerated. A staff reorganization was performed with more personnel added to the staff. Planning and scheduling for the entire project began. Planning shifted from low contamination equipment to higher radioactive areas and components.

3.5.1 Hot Spot Reduction Program

The purpose of a Hot Spot Reduction Program was to identify and remove components or parts of components that were highly impacting area dose rates in certain plant areas. Hot spot reduction would be performed prior to the removal of other equipment in an area. This approach reduced the overall personnel exposure required to perform commodity removals in an area. The Hot Spot Reduction Program at RSNGS was performed in plant areas where all systems had been drained and components with "hot spots" were identified.

A hot spot reduction effort was undertaken to the extent that it could be done without affecting the existing required plant configuration.

3.5.2 Cool and Dim

The control of systems at Rancho Seco was more like that of an operating plant where only limited isolations of systems (i.e. electricity and ventilation) took place as needed. This approach was referred to as "cool and dim." At RSNGS decommissioning was conducted in limited areas on specific components at different times. This was different than the approach taken at other plants where there were major electrical and system isolations referred to as "cold and dark." The "cool and dim" approach was used at Rancho Seco for a number of reasons:

- Plant experienced personnel were involved (operations and electrical.)
- The extended schedule allowed activity to progress a room or area at a time.
- Initially, work could be terminated based on their experience at any time.

3.5.3 Historical Site Assessment

The purpose of a Historical Site Assessment (HSA) is to determine the extent and nature of the contamination at the site by reviewing incidents that occurred during the operation of a plant.

The initial RSNGS characterization was not performed in accordance with the guidelines of the "Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)," because this document did not exist. The HSA effort began in 2000 but was not complete until 2004. In August 2006, Rev. 1 was issued, but it contained only very minor changes. One of the changes was the inclusion of a specific discussion on the lack of on-site radwaste disposal. This specific reference was requested by the NRC.

The methods used to conduct the RSNGS HSA included:

- Computer searches of plant records with a follow-up review of the documents identified. The search targeted radiological incident files, operational survey records, and annual environmental reports to the U.S. NRC.
- Distributed questionnaires to past and present employees asking them to identify past contamination incidents. Response to these questionnaires could be anonymous if so desired by the responder. Follow up interviews of responders were conducted.
- Plant Photos and Plant Modification documentation was reviewed to determine the movement of soil into and around the site. This work proved very useful in locating contaminated areas.

Although the RSNGS HSA focused primarily on the radiological condition of the site, the effort also evaluated hazardous and state-regulated non-radioactive materials at the site that could have required remediation and disposal.

3.5.4 Initial Scoping and Site Characterization

Concurrently the results of the HSA were used along with the characterization surveys to determine the radiological status of site systems, structures and land areas. This significant effort can be summarized in the following major tasks:

- System Status – This effort was concentrated on systems not connected to known contaminated systems or systems that could have been contaminated by leakage from known contaminated systems. Sampling of known contaminated systems was not needed as previous Health Physics routine surveys provided adequate data to make such assessments. Scoping surveys were conducted by opening valve bodies and filters, component manways and in some cases cutting into systems that had been removed from service and drained.
- Structure Surveys – As with systems, this effort concentrated on structures that were not known to be contaminated. Smear survey and samples obtained by concrete coring were used to determine the presence of contamination and the amount of remediation needed.
- Land Areas – Soil sampling was conducted using both manual surface measurements and split spoon equipment. As site specific release limits were being developed as of this time, the sample concentrations were compared to initial values that had been developed. The samples also allowed the estimation of the quantity of soil requiring remediation.

The results of the characterization surveys were compiled in the License Termination Plan (LTP) that also included surveys collected as a part of the Operational Health Physics Program.

The information developed during the initial characterization program defined the radiological and hazardous material assessment based on the knowledge and information available at the end of 2005. Completion of the Initial Site Characterization allowed the RSNGS Decommissioning Project to:

1. Divide the RSNGS site into manageable sections or areas for survey and classification purposes;
2. Identify the potential and known sources of radioactive contamination in systems, on structures, in surface or subsurface soils, and in groundwater;
3. Determine the initial MARSSIM classification of each survey area;
4. Develop the initial radiological and hazardous material information to support decommissioning planning including building decontamination, demolition, and waste disposal;
5. Develop the information to support Final Status Survey design including instrument performance standards and quality requirements; and
6. Identify any unique radiological or hazardous material health and safety issues associated with decommissioning.

Operational radiation surveys and additional characterization measurements and samples obtained during the continuing decommissioning activities were used to confirm the area classification and effectiveness of the cleanup activities before completing the Final Status Survey.

As a result of the HSA and site characterization, all but approximately 93 acres of the plant site were initially identified as “non-impacted” as defined in MARSSIM. The results of ongoing surveys were used to identify areas of the site that require decontamination, as well as to identify the cleanup methods and plan for their associated costs.

It is recommended that a thorough site characterization be performed early in the decommissioning if not prior to permanent shutdown. The results of a site characterization may effect how the decommissioning is conducted. This is an essential step for sites using a Decommissioning Operations Contractor (DOC). However, for Rancho Seco, who did not use a DOC, this was a best practice and not an essential step since they had time to survey throughout the decommissioning process.

4 DECOMMISSIONING SUBCONTRACTORS

4.1 Self-Performance/Decommissioning Support Contractor

SMUD initially intended to select a Decommissioning Operations Contractor (DOC) for decommissioning in 2008 once the decommissioning fund was complete. With this in mind, the initial (post-shutdown) decommissioning organization included a mix of retained RSNGS employees that could maintain the fuel pool operation and prepare for fuel movement. As the time for fuel movement was delayed and incremental decommissioning began there was a need for the actual personnel (craft and field supervisors) and heavy equipment to begin the decommissioning process.

To fill this need, RSNGS issued a request for proposals for a decommissioning support contractor in late 1997. The support contractor needed to be able to provide the necessary personnel and equipment and allow the project to be self-managed by RSNGS. This contract arrangement allowed for decommissioning to be stopped at any time if necessary. After an evaluation process, a decommissioning support contractor was chosen. This arrangement has continued throughout the decommissioning.

4.2 Additional Subcontracts

There were a number of additional subcontracts awarded by RSNGS during the decommissioning. These contracts were generally for specialized services. The specialized subcontractors included projects such as:

- Asbestos Abatement
- Waste Processing and Disposal (performed under several different contracts)
- Movement of Fuel
- Water Processing
- Reactor Internals Segmentation
- Cutting of Primary Components and Large Bore Piping
- Cutting of the Reactor Head
- Movement of Large Components
- Supply of the Reactor Vessel Cutting System
- Removal and Disposal of all Reactor Building Concrete

Experiences under some of the key activities will be discussed later in this report.

After RSNGS decided to self-manage the decommissioning in 1997, staff began contracting for major activities that were expected to begin. Initially, the contracts called for key personnel for planning, dismantlement and waste management.

A disposal contract was placed with Envirocare of Utah. This contract with numerous additions has continued throughout the entire decommissioning. While Envirocare offered a contract to encompass all decommissioning waste (at a much reduced rate) the required provision to be able to stop at any time would not allow this option.

Multiple contracts for waste processing were put in place to achieve the most cost effective solution for each waste batch.

SMUD requires all procurements above a modest level to be competitively bid. It should be noted that procurement requirements are based on legal requirements in the MUD-Act, as well as Board-directed processes. As a public entity, much of the procurement process is dictated by law, not merely the desire of executive management.

At times this process brought cost–effective bids. Other times this process interfered with the schedule and required extensive negotiation and justification to arrive at a successful contract.

5
SPENT FUEL STORAGE

5.1 Selection of Long Term Fuel Storage Approach

An economic evaluation was performed soon after shutdown that compared the long-term cost of two fuel storage options:

- Continued storage of the fuel in the pool inside the fuel building or
- Construction of an Independent Spent Fuel Storage Facility (ISFSI)

The major costs of continued wet storage of the fuel were:

- Operation and Maintenance Personnel to monitor the status of the pool and keep the systems in the fuel building operable
- Maintaining security and radiological protection of the restricted area containing the fuel building
- The decommissioning of nearby buildings would be made more difficult as the safe storage of the fuel would need to be maintained.

The major costs of the ISFSI fuel storage option were:

- Design and construction of the storage pad and other systems for the dry storage casks.
- Fabrication of the 22 concrete storage modules and fuel storage canisters.
- Facilities and personnel to maintain the security of the ISFSI.
- Transfer operations to load the fuel from the pool to the ISFSI pad.
- Activities required for transferring the fuel canisters to the Department of Energy (DOE) for disposal.

The economic evaluation determined that the ISFSI option was the most cost effective and placed the fuel in a more secure configuration for long-term storage.

SMUD decided that a transportable dry cask system was desirable to allow the fuel to be transported to the DOE without replacing it in a fuel pool for repackaging. No such system existed at the time that would accommodate Rancho Seco's fuel. SMUD decided to develop and purchase a "first ever" large-scale canister based transportable spent fuel storage system. Initially the development was conducted as a cooperative, cost-sharing partnership with the

DOE. As time went on, the DOE eventually withdrew from the project, and the entire project was funded by SMUD.

SMUD 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 licensing work. The cost due to the fuel transfer delay was the single largest item which increased the overall decommissioning cost.

The RSNGS 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 fixed structural array and are then seal-welded at both ends. The horizontal storage modules are thick reinforced concrete storage bunkers for the canisters. The twenty-second module was provided for storage of Greater Than Class-C (GTCC) waste from reactor vessel internals segmentation.

5.2 Spent Fuel Pool Island

Once it became clear that dismantlement would continue beyond the incremental phase, the fuel storage interfered with the plant decommissioning for the following reasons:

- The spent fuel was stored in a building that was immediately adjacent to the reactor containment building and the auxiliary building.
- Numerous plant systems that supported the operation of the spent fuel pool were located in other buildings. These included multiple water systems for cooling, liquid radwaste treatment, ventilation and the electrical supply systems.
- Due to the location of the pool near other plant structures, decommissioning activities near to the pool needed to be performed much more carefully to insure the continued safe storage of the fuel.

In the short time, the fuel building was modified to be a stand alone facility not requiring the operation of any systems located in other buildings. This concept involved turning the fuel building into a “nuclear island” which would allow more of the decommissioning activities on site to proceed. The required modifications involved the following:

- Installation of a refrigeration ultimate heat sink system to expel the decay heat from the fuel to the pool to the atmosphere. This modification allowed all other cooling water systems to be removed from service
- Installation of a fuel pool cleanup filter and ion exchanger eliminating the need for the permanently installed fuel pool cleanup system
- Installation of emergency diesel generator to ensure a backup power supply to the fuel pool equipment
- Other modifications to electrical power systems and makeup water supply

- A new equipment qualification and quality standard was established for use on the fuel building island cooling system. It was determined that due to the time required to reach a critical temperature (about 30 days to 140 degrees Fahrenheit.) the system was not required to be classified as a Class 1 safety related system.

These modifications were completed in 1999.

Once the fuel building had been converted to contain all required support systems, the plant systems in other areas could be reclassified for removal and disposal. In particular, declassification of systems that had been contained in the auxiliary building and the tank farm allowed system removal and demolition activities to proceed sooner. Previously, removal of these systems was tied to movement of the fuel to dry cask storage.

5.3 Fuel Transfer to Dry Storage

Fuel movement began in May of 2001. Loading a single canister took approximately a week and a half to two weeks to complete. The schedule was hampered by the delivery rate of the canisters from the fabricator. However all canisters were finally on-site in May 2002 with spent fuel loading and storage completed on August 21, 2002.

Dose rates on the loaded transfer cask were significantly below the projected dose rates bringing the annual site exposure well under the ALARA goals. The transfer cask was electro-polished prior to its first placement in the fuel pool and this made for a quick decontamination process after removal from the pool further lowering the total exposure.

6 REGULATORY AND STAKEHOLDER INTERACTION

6.1 Site Release Criteria

There were a number of regulations and agreements that were factors in determining the site release criteria. The following addresses the interactions with the regulators and other stakeholders in determining the site release criteria while the technical details of the release criteria will be discussed in Section 8.

6.1.1 Radiological Release Criteria

The determination of the radiological release criteria for the Connecticut Yankee and Maine Yankee sites required interaction with the U.S. Nuclear Regulatory Commission (NRC), their respective states, the U.S. Environmental Protection Agency (EPA) and local stakeholders. These plants were located in agreement states that were allowed to establish regulations independent of those of the NRC or the EPA. Each of these interactions provided significant complications in arriving at a mutually agreed upon release criteria. SMUD's interaction was primarily with the NRC since California is not an agreement state. A discussion of how the resulting release limits compare is provided in Chapter 8.

6.1.1.1 U.S. Nuclear Regulatory Commission

NRC Code of Federal Regulations Title 10 Part 20.1402 (Reference 8), defines the standard to which a site to be released for unrestricted use must meet. The regulation requires that no average member of the critical group receive a post closure dose of more than 25 mrem/yr (1.25 mSv/yr). (Total Effective Dose Equivalent-TEDE as will be discussed below.) There is an additional requirement that an evaluation be performed to determine the level of effort or additional remediation that may be necessary to meet the ALARA requirements.

To define how SMUD was to meet this regulation for the decommissioning of Rancho Seco, a License Termination Plan (LTP) was written and submitted to the NRC in April of 2006. This plan included the results of calculations of the remediation limits that needed to be achieved to meet the NRC 25 mrem/yr (0.25 mSv/yr) standard. The NRC accepted the RSNGS LTP as adequate for review in late 2006 and subsequently sent two requests for additional information (RAIs) to RSNGS in October of 2006 and January of 2007. The RAIs were in the general categories of:

- Adequacy of Site Characterization
- Survey Area Classification
- Survey of Subsurface Soils
- Dose Modeling Parameters

After numerous meeting and conference calls, all NRC comments were resolved and the LTP is expected to be approved in November of 2007.

As defined by the NRC LTP approval process, a public meeting was conducted followed by a period where interveners could forward issues. These issues would then be resolved through a hearing process before the NRC Atomic Safety Licensing Board (ASLB). No interveners appeared.

6.1.2 Other Site Release Issues

While other site remediation for hazardous materials has occurred, no regulatory site release is expected because SMUD intends on continued use of the site.

7
USE OF TECHNOLOGY

7.1 Water Management

Rancho Seco receives its water from a canal both for cooling and make-up. Water is discharged to a creek which is dry most of the year if no water is discharged. This provides little dilution for radiological or chemical discharges. This situation provided the initial hurdle for decommissioning to begin. Approximately two million gallons of highly tritiated, borated and radioactive water was on site in various tanks, pools and systems. A water management plan was one of the first key tasks.

Soon after shutdown plant evaporators could process the boric acid and radionuclides into a much smaller volume of concentrates. A blender/dryer could take the concentrates to dry powder for disposal. However, by the time incremental decommissioning began the required systems were no longer operational.

A contract was placed for the use of a vendor supplied reverse osmosis system. This allowed the removal of the boric acid and a sufficient fraction of the radionulides to release the discharge liquid after passing through demineralizers. Downstream dilution for tritium was also required. This still left about 50,000 gallons of concentrates for disposal.

Under an EPRI project, the concentrates were further concentrated by a Vibratory Shear Enhanced Filtration Process (VSEP) system [4]. Use of the VSEP equipped with Reverse Osmosis (RO) type membranes provided significant concentration volume reduction.
This allowed concentrated waste to be dried in drum dryers.

Water management continued to play a major role throughout the decommissioning project. Wastewater continued to be produced from decontamination, concrete cutting, in-leakage, Reactor Vessel (RV) Head Cutting, RV Internals Segmentation project and the RV Segmentation project. Some of these sources produced water with significant processing issues. The RV Segmentation project produced approximately 100,000 gallons of water with poorly filtered garnet fines. This water was processed with a polymer additive and settling following by filtration and demineralization.

7.2 Shipment and Disposal of Primary Components

7.2.1 Reactor Head Disposal

A major work activity during 2003 involved the disposition of the Reactor. This Babcock and Wilcox design consisted of sixty-nine Control Rod Drive Mechanisms (CRDMs), each weighing approximately 1000 pounds (453.6 kg); a Service Structure weighing 35,000 pounds (15,875 kg) and the Reactor Head itself, weighing 160,000 pounds (72,575 kg.)

This work began with removal of the Service Structure from its mounting on the Reactor Head after abating lead-based coatings and flame cutting the lower shroud from the Head. The Structure was removed from the Refueling Cavity and taken to an adjacent work area where it was segmented for disposal. These sections were packaged into a 20 foot Seavan and were subsequently sent to a processor for decontamination and free release or further volume reduction.

The next step was to remove the CRDMs from the Reactor Head. The CRDMs were grouped by their applicable function during plant operation and consisted of safeties, control and power shaping rods. There was very little radiological data associated with the CRDMs and lead-screws, which connected directly to the control rods. Lack of data resulted in the dismantlement crew proceeding very carefully during CRDM removal. Packaging criteria for the CRDMs was established based on the activation and associated dose rates and burial site waste acceptance criteria.

Surveys of the first CRDM removed indicated low dose rates. The survey along the length of the lead screw indicated 50 to 60 mrem/hr (0.5 to 0.6 mSv/hr) gamma while the tip of the lead screw was 40 mrem/hr (0.4 mSv/hr) gamma. There was little fluctuation in dose rates as the different CRDM groups were removed and surveyed. The CRDMs were mounted to the Reactor Head and were removed by cutting the nozzles just below the mounting flange by use of a Tri-Tool clamshell, which was fitted around the nozzle. Once cut, the CRDM was lifted from the cavity, surveyed and placed in a processing area where it was segmented into box-sized lengths for disposition. All were segmented and packaged within a metal strong tight container (STC) and sent for direct disposal.

During the first part of 2003, a Request For Proposal (RFP) was issued for transportation and disposal of the Reactor Head and Pressurizer to Envirocare of Utah. In addition RSNGS negotiated a disposal rate for the intact Head and for an option of five sections of the Head. The Reactor Head to be transported was described without the Service Structure and CRDMs. It became apparent that it would be most cost effective to segment the Reactor Head prior to disposal. The pricing for the disposition of the Reactor Head considered engineering and fabrication of a suitable container for the entire Reactor Head and the final disposal cost. This was compared to the costs of the disposition of a segmented Reactor Head. The segmented pieces would have low source terms and dose rates and would allow easy handling and packaging into 20 foot open-top Seavans. The Seavans would serve as strong tight containers. The remaining cost would be transportation cost and disposal at the Envirocare of Utah facility.

Rancho Seco's cost for segmentation and shipping was half the estimated cost for intact Reactor Head disposal.

The Reactor Head was segmented with a diamond wire rope supplied by segmentation vendor Bluegrass. The five segmented sections included three sections of the flange and two sections of the top portion of the Head, cut just off-center through a clear path around the remainder of the CRDM nozzles.

Contact dose rates underneath the Head were 200 mrem/hr (2 mSv/hr) and dropped off to 80 mrem/hr (0.8 mSv/hr) at the open plane of the bottom flange. Dose rates on the exterior of the Head ranged from 15 to 30 mrem/hr (0.15 to 0.3 mSv/hr.) The four flange keyways had contact dose rates up to 800 mrem/hr (8 mSv/hr) and required shielding as the segmented pieces were prepared for shipment. High levels of contamination were found on the underside of the Head and were affixed with use of a polymer-based latex paint.

Figure 7-1
Reactor Head With First Piece Removed

7.2.2 Reactor Coolant Pumps and Pressurizer

The removal and shipment of the reactor coolant pumps and the pressurizer was fairly straightforward and accomplished without incident. Due to the relatively low activity levels of these components, shipment was by rail to the Envirocare facility.

7.2.3 Steam Generator Removal, Shipment, and Disposal

Rancho Seco's Steam Generators (SG) are of Babcock &Wilcox (B&W) design and commonly known as Once-Through Steam Generators (OTSG). The B&W design consists of two such SGs, each approximately 80 feet (24.4 m) in height, 12 feet (3.7 m) in diameter, and over 550 tons (500 metric tons) in weight. The SGs were too large to ship to Envirocare in their intact state due to rail clearances with respect to the length of the generator and certain radii of track along the required route. Therefore Rancho Seco cut the SGs in the latitudinal direction at approximately the halfway point and then capped the exposed cuts with large steel plates to meet rail and Department of Transportation (DOT) requirements. This enabled the SGs to be shipped directly for disposal to Envirocare.

7.2.3.1 Transportation Evaluation

In the fall of 2001, the first step taken in planning the disposition of the SGs involved a railroad transportation evaluation to ascertain the feasibility of available routes from Rancho Seco to Envirocare. Duratek Services, Inc, was contracted to determine acceptable shipment configurations and the feasibility of available routes. Ultimately the decision to segment each SG into two sections was made. This decision was based on:

1. The ability to clear the shipment route with the segmented SG, and
2. Previous success Rancho Seco had in segmenting the Reactor Head and,
3. Transporting other components via rail utilizing MHF-Logistical Solutions (MHF-LS), a company contracted by SMUD to coordinate radwaste transportation logistics.

Rancho Seco worked with MHF-LS to ascertain acceptability for railroad routing and clearance between Sacramento and Envirocare. The shipment would consist of each SG section positioned on a 12 axle QTTX series 131627 – 131636 railcar NSH53 class with a load limit of 743,000 pounds (337,000 kg) and light railcar weight of 202,000 pounds (91,626 kg.) The 1.5 inch (3.81 cm) deck plate is 53 feet (16 m) in length and 10 feet 8 inches (3.2 m) wide. Each shipment would consist of an upper and lower section of SG on separate railcars.

Once the rail clearance and routing was established for the SG sections, Rancho Seco proceeded with the request for regulatory exemption from the Department of Transportation.

7.2.3.2 Segmentation of the Steam Generators

The SG project began in the 3rd quarter 2004. An early activity involved decontamination of the SGs to a level below 20,000 dpm[4]/100cm^2 (beta/gamma). Pressure washers were utilized for the decontamination and proved to be quite effective in reducing loose contamination levels to well below 20,000 dpm/100cm^2.

The SG segmentation scope of work was to include a latitudinal cut across the SG tubes at approximately the halfway point. To facilitate access to the tubes, Rancho Seco cut four windows in the steam generator carbon steel housing. Each window was approximately 6 inch (15 cm) high by 4 feet (1.2 m) in length. The dose rates at the open plane to the outer window

[4] dpm = disintegrations per minute

openings were 100 mrem/hr (1 mSv/hr) and dropped to 30 mrem/hr (0.3 mSv/hr) at 12 feet (3.7 m) from the open window. Contact dose rates to the outer diameter to the tubes bundles were 300 to 400 mrem/hr (3 to 4 mSv/hr.)

The SG segmentation contract was awarded to diamond wire cut these components. Rancho Seco had been successful in with this cutting method in segmenting the Reactor Head in late 2003. The contact included removal of the cold leg nozzles at the bottom of each SG.

The diamond wire sawing system passed through the windows cut in the SG walls. The diamond wire was then coupled together with steel sleeves and passed around a drive wheel and over guide pulleys. Wire tension was maintained by a rack and pinion system that moved the drive wheel along the wire saw carriage. The main drive wheel had a “V” groove and rubber lining to grip the wire. A hydraulic motor rotated the drive wheel at a controlled rate and direction selected by the operator.

Work began in August of 2004 by trimming the cold leg nozzles from each SG to meet the envelope diameter dimension cleared for rail transport. By the end of August successful latitudinal cuts were completed across the SG tubes. No water was present in the tubes. With two men working on each SG the segmentation was performed in approximately 60 hours. The average cut rate was approximately 270 in^2/hr (0.17 m^2/hr.) Each SG was segmented using approximately 0.2 man-rem (0.002 person-Sv.)

Figure 7-2
Diamond Wire Saw Cutting Tubes Through Window in OTSG Shell

7.2.3.3 Removal of the Steam Generator Sections from the Reactor Building

As the SG tubes were segmented the following tasks were completed in preparation of the removal of the SG sections from the reactor building:

- Plates and plugs were welded over nozzle and pipe penetrations.
- Bolts on flanges were tightened to required torque values.
- Calculations were performed to assure the nozzle covers would provide adequate shielding to meet Department of Transportation (DOT) radiation limits as described in 49 CFR 173.441.
- Trunions were positioned at 180°.
- Tailing lugs was welded onto each section of SG.

The final step was to perform a latitudinal cut around the SG carbon steel housing.

The rigging contractor had assembled lifting equipment on the Rancho Seco polar crane. When the latitudinal cut on the first SG was completed, the top section of the SG was transferred to the Reactor Building hatch. The lift was conducted first in a vertical direction in order to clear the walls surrounding the SG section that extended to an elevation of +60 feet (18 m) above grade level. Once above the +60 foot (18 m) level wall, the SG section was transported in a horizontal direction until it was situated adjacent to the Reactor Building Hatch where it was lowered onto a shield cap. The shield cap, 13 feet 6 inches (4.1 m) in diameter, was designed and fabricated to provide closure over the latitudinal SG cuts.

Once the 4 inch (10.16 cm) thick shield cap was welded to close the latitudinal cuts and two coats of the modified acrylic latex paint PBS, was applied to fix the existing loose contamination to a non-detectable level. At this point the top section of the SG was again lifted using the tailing lug to a horizontal position (Figure 7-3) and lowered onto a multi-axled Goldhoffer trailer for transport out of the Reactor Building for final shipment.

The lower section of the SG remained in place and the 4 inch (10.16 cm) thick shield cap was flown and positioned over the cut. The shield cap was welded in place with trunions positioned at 180° and a tailing lug welded onto the lower section of B SG to then level the section in preparation to set upon the Goldhoffer trailer and removed from the reactor building.

Once out of the Reactor Building the two sections of the SG were transported to the rail spur via the Goldhoffer trailer where they were placed upon the QTTX cars and final preparations were made for shipment.

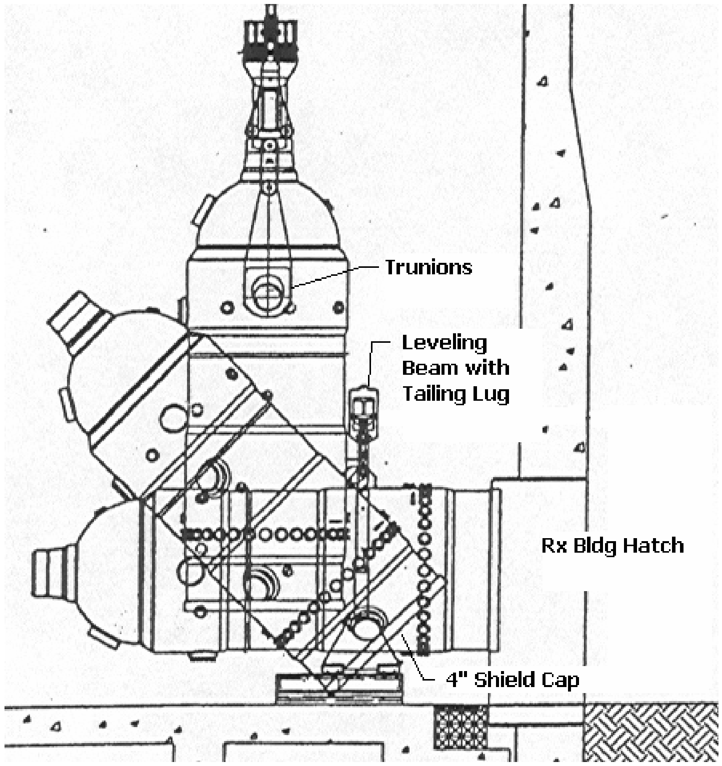

Figure 7-3
Top B SG Section Down-Ending With Shield Cap in Place

7.2.3.4 Shipment to Envirocare of Utah

Once loaded and prepared for shipment, Union Pacific Railroad (UP), MHF-LS, and Envirocare personnel conducted inspections. The UP inspector and MHF-LS assured the blocking and bracing was consistent with the submitted drawings and the actual dimensions (width and height) were within the envelope dimensions as cleared for transport to Envirocare. UP released the cars for shipment on November 23, 2004. Final radiological surveys were performed and no detectable loose contamination was found on either OTSG section.

Contact dose rates on the lower section of the SG ranged from 10 mrem/hr (0.1 mSv/hr) along the top and sides to a single contact dose rate of 100 mrem/hr (1 mSv/hr) located on the cold leg cover shield. The two-inch (5 cm) thick shield had a four-inch penetration that was covered by a ½ inch (1.3 cm) plate which was then removed at Envirocare to permit the introduction of grout. The highest one-meter dose rate, from the penetration, was 10 mrem/hr (0.1 mSv/hr.) All two-meter dose rates were less than 2 mrem/hr (0.02 mSv/hr.) The contact dose rate to the four-inch (10.16 cm) shield cap was 7 mrem/hr (0.07 mSv/hr.) No additional shielding was necessary to meet DOT radiation limits. All dose rates were well within 49 CFR 173.441 radiation limits and no further shielding was required.

The SG sections were offloaded and placed into the appropriate disposal cell locations during the week of December 13, 2004. The QTTX railcars were then surveyed, released and sent back to Rancho Seco in preparation for the second set of SG sections.

Figure 7-4
Steam Generator Sections Leaving Rancho Seco

7.3 Reactor Internals Segmentation

The Reactor Internals Segmentation project began in July 2004 and was completed in May 2006. Based on the geographic location of the plant being 30 miles (48 km) from a navigable waterway and concerns associated with barging an intact vessel, shipment of the Reactor Vessel intact was deemed not practical, and the decision was made to perform 100% segmentation of the Reactor Internals. This would then be followed by segmentation of the Reactor Vessel which would allow overland transport of the segmented portions.

SMUD is currently storing the Level B&C waste from the segmentation of the Internals in special liners designed to be transported to a disposal in Chem Nuclear 8-120B casks. The GTCC material is stored in one special container in the onsite ISFSI. The GTCC canister is functionally identical to the fuel canisters.

The limited power operation of the plant in its operating history and the longer decay time until the internals segmentation activities resulted in a total internals activity at Rancho Seco of just over 73,000 curies (2.7 E15 Bq) at the start of the Internals Segmentation Project.

7.3.1 Segmentation Planning and Vendor Selection

A Request for Proposal was issued in September 2003 to perform the reactor vessel internals segmentation. Rancho Seco plant staff assumed responsibility for radiation protection, waste packaging and some limited support with the onsite maintenance facility and personnel. The level of direct plant support was greater than that for other segmentation projects. This is consistent with the approach that Rancho Seco has employed in decommissioning activities to date.

As a result of the bid evaluation process, SMUD selected a team of 3 organizations, Trans Nuclear, Duratek and Mota, to segment and support segmentation of the internals utilizing the mechanical methods of sawing and milling. This was the first large commercial plant performing internals segmentation strictly using these mechanical methods. The methods used at other plants such as plasma arc, Abrasive Water Jet (AWJ), Electrical Discharge Machining (EDM) and Metal Disintegration Machining (MDM) were not employed. Difficulty at previous plants with the process of containing, collecting and processing the small particles (dross in the case of plasma arc and swarf in the case of AWJ) was a significant factor in the decision. Rancho Seco selected to use the more conventional mechanical sawing and milling methods to segment the internals. As indicated above, the lower curie inventory allowed some flexibility not really available to the other reactor internals segmentation projects with significantly higher activity.

7.3.2 Tooling

Due to the many and varied shaped and thicknesses of the internals, the use of mechanical methods of sawing and milling for internals segmentation required a number of different tooling setups. Typically a robotically delivered plasma torch or AWJ end effecter is somewhat more versatile since the end effecter is delivered by a multi axis manipulator.

The original tooling designed for the project included the following:

- The Reciprocating Machine Tool, RMT (shown conceptually in Figure 7-5.) This tool is guillotine saw. It is like a broach with an 18 foot (5.5 m) long “blade”.
- The 38i, a 38 inch (1 m) diameter carbide tipped saw mounted on one of the towers of the RMT was used to yield vertical slices of the plenum cylinder, core barrel, and thermal shield.
- The Circumferential Hydraulically Operated Cutting Equipment, C-HORCE (shown conceptually in Figure 7-6) is a internal mounted milling machine. It was designed to make circumferential cuts. It was used to cut cylinders from the core barrel and thermal shield.
- The Bolt Milling Tool, BMT (shown conceptually in Figure 7-7) was designed to mill out bolt heads. It is mounted on a track that is clamped to the work piece. It employed an end mill as the cutting tool for removing the heads of bolts in the baffle structure.
- The Bolt Shearing Tool, BST, was designed to shear the GTCC formers holding the former plates in position.

- The Machine Chip Collection System, MCCS, was designed to collect the machining chips for packaging and disposal.
- Cavity water filtration and purification was performed using an Advanced Liquid Processing System (ALPS). This system was augmented with a Hign Integrity Container system for backwashes and 2 10,000 gallon (37,850 liter) tanks for ultimate cavity water release.
- The 8-120 shield transfer bell supplied by the waste vendor designed with 2 spray rings to decontaminate the exterior surfaces of the special 8-120 liners which were loaded wet in the cavity with segmented internals. The shield bell incorporates 4.5 inches (11.43 cm) of lead. It is positioned over the special liners at the cavity surface and utilizes it own hoist to lift the liners from the cavity floor into the shield bell to transfer to another shielded location.

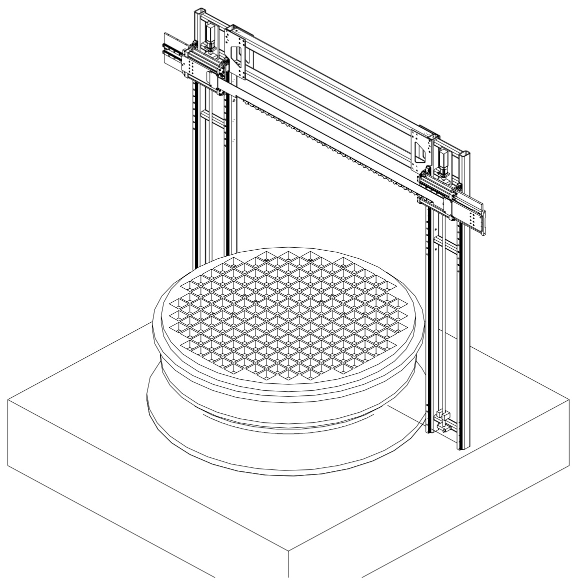

Figure 7-5
Conceptual Sketch of Reciprocating Machine Tool, RMT

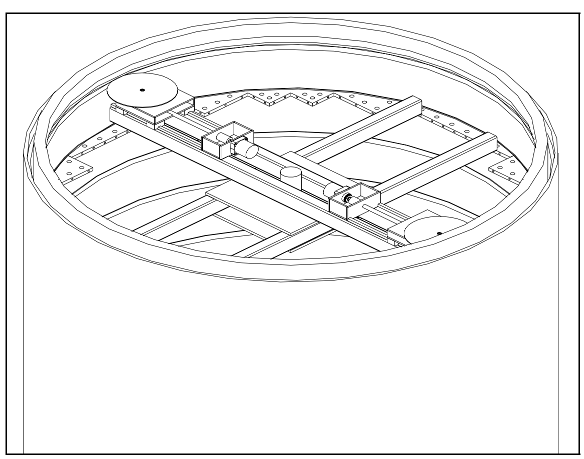

Figure 7-6
Circumferential Hydraulically Operated Cutting Equipment (C-HORCE)

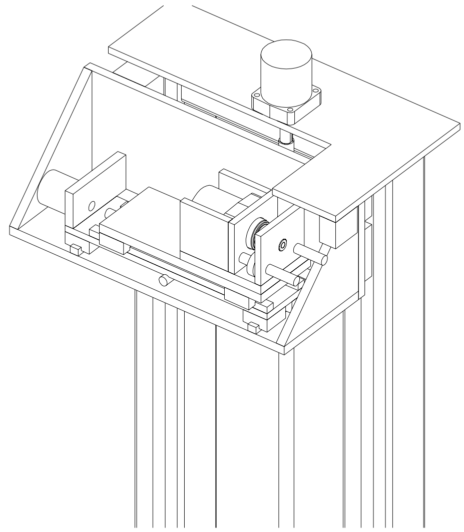

Figure 7-7
Conceptual Sketch of Bolt Milling Tool, BMT

In addition to the original tooling, the following was also used in an unplanned cutting of the Plenum Assembly:

- A hand held Plasma Torch was not part of the original tooling planned and was used with a tent to cut the plenum assembly. As would be expected, this unplanned activity added to the originally planned personnel exposure.
- A diamond wire saw was used to make initial cuts of the plenum cutting off the hotter lower end (upper grid) and used to split the plenum in lengthwise segments. This was performed remotely in air.
- An exterior pole saw was used to cut out the guide tubes.

7.3.3 Tool Testing

Testing of the Segmentation Tooling was performed by the vendors. Although some mockups of portions of the internals were built and cut with the tooling, the testing was primarily a tool testing program and was not a "high fidelity" mockup testing program. Since so many specialized tools are required for mechanical cutting, an aggressive testing program could tend to be more involved than with AWJ or Plasma testing methods. For example since rigidity is very critical to mechanical sawing or milling, the test piece and tool need to accurately reflect the in plant rigidity. Also sawing operations, particularly with the 18 foot (5.5 m) long blade with the RMT, failed to represent the field arrangement and accurately reflect needed rigidity. The mockups did not reflect the same combined thicknesses as the actual components being cut nor the same cutting tool and clamping mechanism (See Figure 7-8).

For example, as discussed in the next section, the test program was not aggressive enough to support the successful implementation of the RMT or the cleanup systems in the actual field application.

Figure 7-8
Reciprocating Machine Tool Mock-Up Testing

7.3.4 Field Experience

Field mobilization of the equipment commenced in the Spring 2005. Some of the key experiences and challenges included:

- The duration of the actual cutting portion of the project took slightly longer than planned, but was completed in approximately 26 weeks.
- All of the cutting equipment, other than the RMT, functioned acceptably with only minor debugging.
- The experience with the cleanup and chip collection systems was less than satisfactory.
- The cutting of the plenum with RMT had to be halted repeatedly due to breakage of cutting blades.
- The segmentation of the plenum had to be performed in air using hand-held plasma arc and saws due to the failure of the RMT.
- The plenum segmentation became a significant dose contributor to the project because of the change in cutting methods.
- Overall the project was performed in less time than and at less cost than other internals projects in spite of the increased cutting required to segment the entire vessel internals package.

More details related to the Rancho Seco Reactor Internal Segmentation Project can be found in *Reactor Internals Segmentation Experience Report: Detailed Experiences 1993-2006* (EPRI Report 1015122, 2007.)

7.4 Reactor Vessel Segmentation and Shipping

7.4.1 Transportation Evaluation

In the fall of 2001, the first step taken in planning the disposition of the Reactor Vessel and Internals involved a railroad transportation evaluation to ascertain the feasibility of available routes from Rancho Seco to the Barnwell Waste Management Facility or Envirocare, Utah disposal location. Duratek Services, Inc, was contracted to determine acceptable shipment configurations and the feasibility of available routes. The three routes Duratek considered for transport of the 38 feet (11.6 m) long, 18 feet 6 inch (5.6 m) diameter (envelop diameter after packaging), 500 ton (454 metric ton) vessel proved unsuitable for intact vessel shipment. Around this time, SMUD management had made the decision not to use the Barnwell facility since SMUD had never shipped waste to that facility. This decision was a key factor in the planning as well as the decision making process since only the Envirocare facility could be used and the B&C internals would need to be packaged for on-site storage. Rancho Seco's only shipping option was to segment the RV and the Internals and transport the waste section packages by rail to the Envirocare facility.

7.4.2 Evaluation of Available Segmentation Processes

When evaluating available segmentation processes, Rancho Seco considered such parameters as cutting speed, secondary waste generation, vapor generation, accuracy, depth of cut, and experience. Mechanical processes evaluated include mechanical/reciprocating saw, rotary saw, and diamond wire. These processes were not selected due to ALARA concerns, i.e. workers close in proximity to the RV; the inability to pierce the metal and an anticipated slower cutting speed.

Thermal cutting processes evaluated included plasma, oxygen and carbon arc; oxy-fuel gas; and oxygen lance. Due to ALARA concerns, questionable control of airborne contamination and generation of Hexavalent Chromium during cut-up of stainless steel; these processes were not selected.

Ultimately, a robotically controlled abrasive water jet was selected as the segmentation could be conducted remotely reducing worker exposure; was cost effective, could be used to pierce holes through the metal and had a production rate which could meet our schedule. The drawback of the abrasive water jet system was the secondary waste that it created, a wet garnet material, which resulted in a fairly low activity waste stream although it proved to be a challenge for collection and subsequent dewatering/drying.

7.4.3 Segmentation Equipment Description

7.4.3.1 System Configuration

The S.A. Robotics system selected for the dismantlement of the Rancho Seco reactor consists of a six-degree-of-freedom robotic arm (Manipulator) mounted on a mast centered inside the reactor vessel. The Manipulator holds a waterjet cutting head which delivers high pressure water at 50,000 psi (3515 atm.) Garnet is mixed with the water at the cutting head to create an extremely abrasive jet stream which will cut through the massive vessel sections (3 inch to 12 inch, 7.6 cm to 30.5 cm.) The Manipulator is capable of ± ¼ inch (0.635 cm) accuracy and ± 1/16 inch (0.16 cm) repeatability at its furthest extensions and through all range of required motion.

The center mast is supported by a polar gantry above the reactor and rests on the bottom of the vessel. The polar gantry allows for the removal of the cut sections of the vessel by rotating the section out of the way. A second gantry placed on top of the first holds the retrieval arm that travels outside of the vessel. Waste from the cutting process is collected by this retrieval arm and sent to a process skid for the waste processing. Waste can also be collected by vacuum on the cutting head during segmenting, and the waste was not otherwise collected then removed by a vacuum hose at the bottom of the vessel.

7.4.4 Segmentation

The RV was to be segmented into 21 sections (Table 7-5 and Figure 7-10). Three flanges, six hot/cold leg nozzles, 2 core flood nozzles, six beltline sections, three hemispherical lower bowl sections and one bottom bowl section. The precise operation of abrasive water jet system ensured actual cuts were very close to actual cuts in overall size and weight.

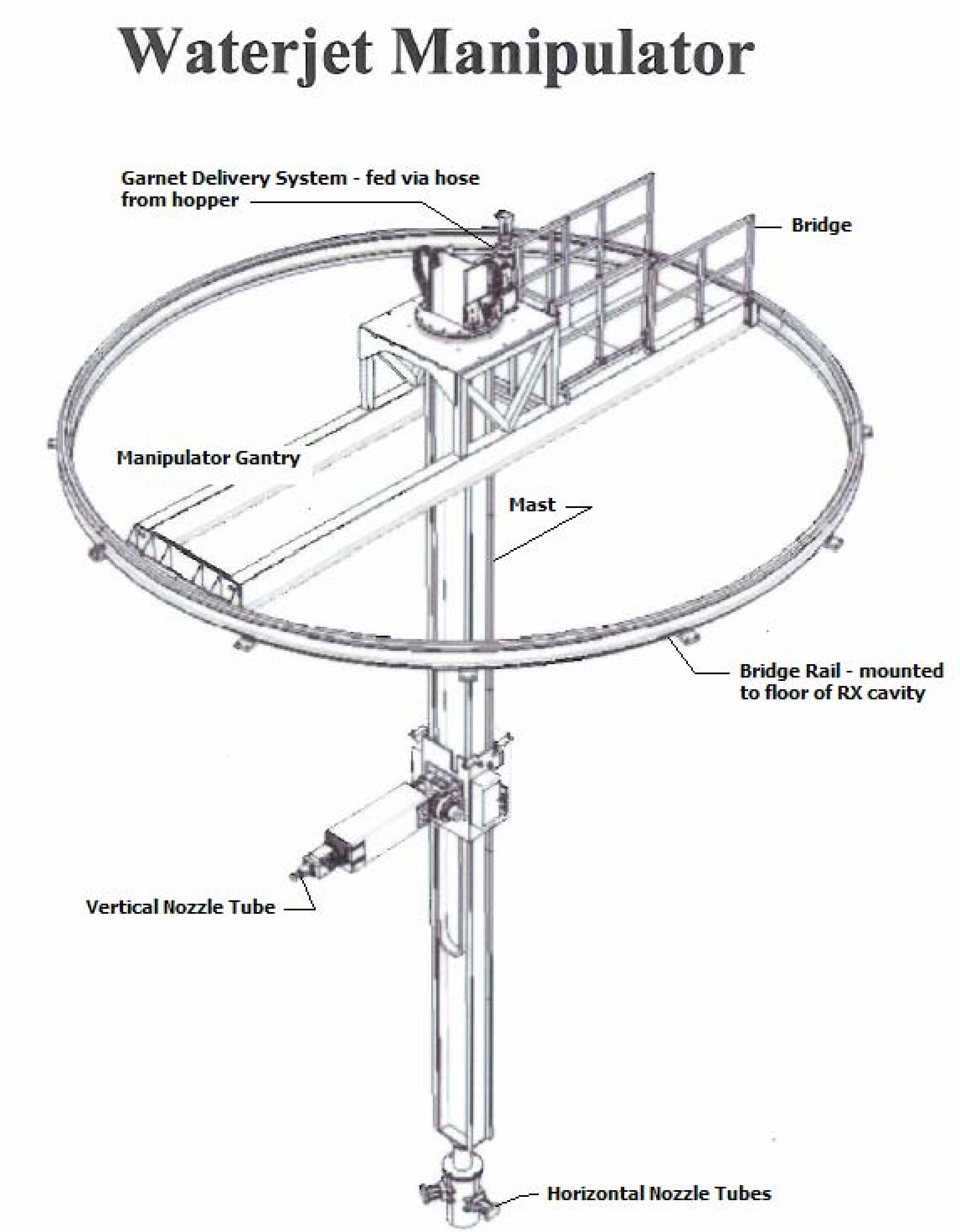

Figure 7-9
Abrasive Water Jet Manipulator

Table 7-1
Reactor Vessel Sections Physical Data

Section	Quantity	Section Weight (Pounds) [1 Pound = 45 kg]	Density Volume (@ 500 Pounds/ft^3) [1 ft^3 = 0.03 m^3]	Total Weight (Pounds) [1 Pound = 45 kg]
Flange	3	25000	50 ft3	75000
Cold Leg Nozzle	4	33400	66.8 ft3	133600
Hot Leg Nozzle	2	33400	66.8 ft3	66800
Core Flood Nozzles	2	13500	27 ft3	27000
Beltline	6	34500	69 ft3	207000
Hemi Head Lower Bowl	3	35000	70 ft3	105000
Hemi Head Center Bowl	1	10000	20 ft3	10000
Totals	21			624400

7.4.4.1 Reactor Vessel Flange

The three sections of RV flange each weighed approximately 25,000 pounds (11,340 kg.) Each section was arc-shaped and was approximately 12 feet (3.7 m) in length by 2 feet (0.6 m) high and 2 feet (0.6 m) wide. The highest contact reading obtained was 250 mrem/hr (2.5 mSv/hr) which was taken adjacent to alignment guides on the flange for placement of the upper plenum. In general, one foot dose rates along the inner diameter were 40 to 50 mrem/hr (0.4 to 0.5 mSv/hr.) The exterior one-foot dose rates were 5 to 10 mrem/hr. The average one-foot dose rate determined for the flange sections was 22 mrem/hr (0.22 mSv/hr) and its corresponding concentration was 0.092 uCi/cm^3 (3.4E3 Bq/cm^3) of Co 60.

7.4.4.2 Nozzles

The six sections of RV cold and hot leg nozzles each weighed approximately 33,400 pounds (15,150 kg.) Each section comprised 60º of RV and was approximately 8- 9 feet (2.4 – 2.7 m) high and 11 inches (28 cm) thick. All sections also had the remnants of the nozzle sticking about 2 feet (0.6 m) out from its side. The highest contact reading obtained was 250 mrem/hr (2.5 mSv/hr) which was taken on the inner diameter, near the bottom of the piece (nearest the beltline region). In general, one foot dose rates along the inner diameter were 100 mrem/hr (1 mSv/hr.) The exterior one-foot dose rates were 15 to 30 mrem/hr (0.15 to 0.3 mSv/hr.) The average one-foot dose rate determined for the nozzle sections was 38 mrem/hr (0.38 mSv/hr) and its corresponding concentration was 0.2 uCi/cm^3 (7.4E3 Bq/cm^3) of Co 60.

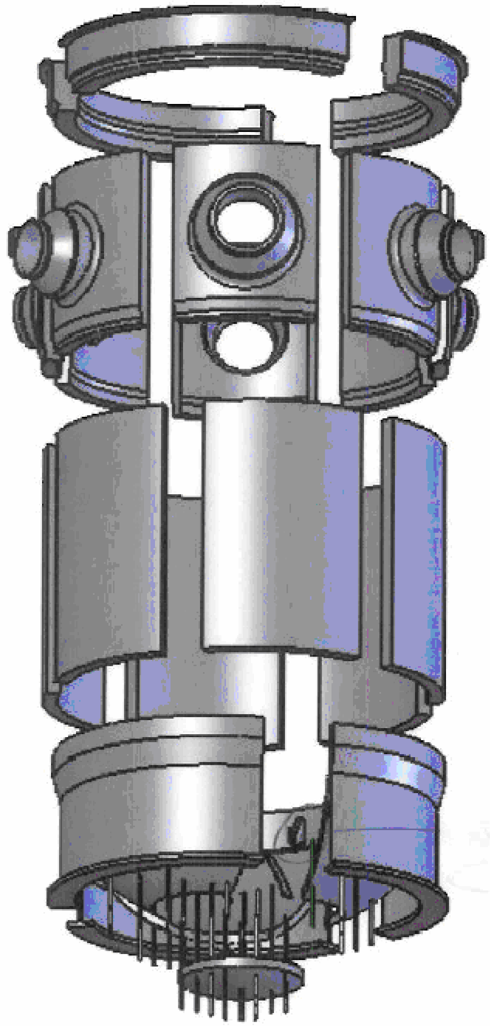

Figure 7-10
Reactor Vessel Segmentation Plan

The two sections of RV core flood nozzles each weighed approximately 13,500 pounds (6,124 kg.) Each section comprised about a 3 feet (0.9 m) wide section of RV and was approximately 8- 9 feet (2.4 – 2.7 m) high and 11 inches (28 cm) thick. The sections also had the remnants of the core flood nozzle sticking about 1 foot (30.5 cm) out from its side. The highest contact reading taken on the inside of the nozzle was 300 mrem/hr (3 mSv/hr.) In general, one foot dose rates along the inner diameter were 100 – 120 mrem/hr (1 – 1.2 mSv/hr.) The exterior one-foot dose rates were 10 mrem/hr (0.1 mSv/hr.) The average one-foot dose rate determined for the nozzle sections was 55 mrem/hr (0.55 mSv/hr) and its corresponding concentration was 0.22 uCi/cm^3 (8.14E3 Bq/cm^3) of Co 60.

7.4.4.3 Beltline

The six sections of RV beltline each weighed approximately 34,500 pounds (15,650 kg.) Each section comprised 60º of RV and was approximately 12 feet 6 inches high (3.8 m) and 9 inch (23 cm) thick. The highest contact reading obtained was 2500 mrem/hr (25 mSv/hr) which was taken on the inner diameter, near the middle of the piece. In general, one foot dose rates along the inner diameter were 1000 to 2000 mrem/hr (10 to 20 mSv/hr.) The exterior one-foot dose rates were 30 to 50 mrem/hr (0.3 to 0.5 mSv/hr.) The average one-foot dose rate determined for the beltline sections was 603 mrem/hr (6.03 mSv/hr) and its corresponding concentration was 0.98 uCi/cm^3 (3.63E4 Bq/cm^3) of Co 60. The estimated average one-foot dose rate and corresponding concentration derived in May 2003 was 580 mrem/hr (5.8 mSv/hr) 0.94 uCi/cm^3 (3.45E4 Bq/cm^3) of Co 60.

7.4.4.4 Hemi Head

The three sections of RV hemi head each weighed approximately 35,000 pounds (15,875 kg.) Each section comprised 120º of lower RV bowl and was approximately 8 feet (2.4 m) high 9 inches (23 cm) thick. The highest contact reading obtained was 500 mrem/hr (5 mSv/hr) which was taken on “J-hooks” near the top of the bowl where the lower internals would rest when in place. In general, one foot dose rates along the inner diameter were 100 to 150 mrem/hr (1 to 1.5 mSv/hr.) The exterior one-foot dose rates were 10 to 20 mrem/hr (0.1 to 0.2 mSv/hr.) The average one-foot dose rate determined for the hemi head sections was 85 mrem/hr (0.85 mSv/hr) and its corresponding concentration was 0.14 uCi/cm^3 (5.18E3 Bq/cm^3) of Co 60.

In all cases, all sections were verified to be waste Class A and met requirements for Low Specific Activity (LSA) II, and all subsequent packaging contained less than an A2 quantity of radionuclides hence could be shipped in excepted packaging per 49 CFR 173.427 (b)(4).

7.4.5 Major Milestones

The major milestones of the Rancho Seco Reactor Vessel Segmentation Project were as follows:

- Equipment fabricated, tested and shipped to Rancho Seco in May of 2006
- Receipt of equipment May - June of 2006
- Set-up and training at Rancho Seco June – July 2006
- Cut (2) 10 inch (25.4 cm) core flood nozzle lines July 2006
- Segment (3) Flange sections Aug – Sept 2006
- Segment (8) Nozzle sections Sept – Oct 2006
- Segment Beltline sections Oct – Nov 2006
- Segment Lower Bowl sections Nov – end of Dec 2006

Project delays were caused by the following:

- Garnet and water mixture rocking up during off hours – the solution was to ensure all mixture was cleared from the hoses prior to shutdown
- Slower than anticipated cut rate in the beginning

The last cut was completed on February 22, 2007. Immediate disassembly of the equipment commenced with all equipment considered to be waste.

Rancho Seco was successful in segmenting, removing, transporting, and disposing of the Reactor Vessel from a monetary, ALARA, and safety perspective. The schedule did fall behind by approximately two months due to problems during the initial equipment start up and with garnet collection.

The total cost for RV segmentation, removal, transportation, and disposal was approximately $5,100,000 (3.4 Million Euros, 2.5 Million GBP). The total worker exposure for the project was 6.4 man-rem (0.064 person-Sv).

Advantages in Rancho Seco’s favor to a successful project included:

- A low source term from limited operation
- Long decay period (approximately 17 years)
- Incorporating lessons learned from other facilities, and
- Working with quality vendors to bring together the best possible team to accomplish the job.
- A willingness to do something no one else has done.

The Rancho Seco Reactor Vessel Segmentation project will be the subject of a detailed EPRI experience report in 2008 [5].

Table 7-2
Abrasive Water Jet Performance Estimates

Abrasive Water Jet Cutting Time & Material Usage								
Vessel Cut	**Thickness (Inch) [1 Inch = 2.54 cm]**	**Length (Inch)**	**Ave. Speed (Inch/Minute)**	**Time to Cut (Minute)**	**Water Rate (Gallon/Minute) [1 Gallon = 3.79 Liter]**	**Garnet Rate (Pound/Minute) [1 Pound = 0.45 kg]**	**Total Water Used (Gallon)**	**Total Garnet Used (Pound)**
Circumferential Cut Below Flange	12”	603	0.10	6,030	2.00	1.50	12,060	9,045
Circumferential Cut Below Nozzles	8-7/8”	588	0.20	3,015	2.00	1.50	6,030	4,523
Circumferential Cut Below Belt Region	8-7/16”	588	0.20	3,015	2.00	1.50	6,030	4,523
Bowl Half Cut	5”	145	0.25	580	2.00	1.50	1,160	870
Longitudinal Cuts from Below Flange to Below Belt Region	8-7/16”	1,605	0.20	8,230	2.00	1.50	16,462	12,346
36” Nozzle Cut	3”	251	0.40	628	2.00	1.50	1,255	941
24” Nozzle Cut	3”	452	0.40	1,130	2.00	1.50	2,260	1,695
Total Cut Time	377.15 hr.							
Total Water Used	45,258 gal.							
Total Garnet Used	33,943 pounds							
Number of 55 gal. Drums of Garnet	46.16							

8
SITE RELEASE ISSUES

8.1 License Termination Plan Issues

The municipal utility owners of the Rancho Seco site intend to retain ownership of the site. This allowed Rancho Seco to model restricted future uses of the site using less conservative assumptions in dose modeling. As some utilities (particularly international utilities) do have not have access to low cost disposal options for very low level radioactive waste that are available in the U.S., the approach used by Rancho Seco may offer cost and disposal site capacity savings. Most of the information in this section was obtained from References [6] through [10].

The LTP is expected to be approved in November 2007. Because Rancho Seco used the lessons learned at other facilities to produce the LTP, this will be the first "Revision 0" LTP from a power reactor that has been approved by the NRC.

8.1.1 Industrial Worker Scenario Justification

Rancho Seco has no plans to release any of the Sacramento Municipal Utility District (SMUD)-owned and -controlled 2,480 acre site for public ownership. This allowed the average member of the critical group to be defined as an "industrial worker" defined as a District employee or contractor who is allowed occupational access to impacted areas of the site over the course of his/her employment. The assumption was made that occupancy would be limited to a 50-week year (45 hours per week). It was further assumed that the industrial worker would spend half of his/her time indoors and half outdoors while onsite. This justification applies to evaluating exposure to contaminated building surfaces, surface soils, subsurface soils, bulk materials and embedded piping.

8.1.2 Building Surface and Concrete DCGLs

The primary difference in the dose modeling for concrete at Rancho Seco was the fact that the buildings will be left standing after license termination. This factor allowed certain areas of the plant to be considered inaccessible to workers. The best example of this is upper portion of the containment building liner. As this area will not be accessible without a lift or the construction of scaffolding, different assumptions were used in the Derived Concentration Guidelines Level (DCGL) calculations.

8.1.3 Critical Group for Structural Surface Exposure

The average member of the critical group is defined by Rancho Seco as a District employee or contractor who is assumed to be on-site for 45 hours per week per NUREG/CR-5512, Volume 3.

RESRAD-BUILD Version 3.3 (Released Summer of 2005 by Argonne National Laboratory (ANL)) was chosen as the computational method to calculate structural surface DCGLs. RESRAD-BUILD as used at Rancho Seco considered seven exposure pathways:

- External exposure directly from the source,
- External exposure to materials deposited on the floor,
- External exposure due to air submersion,
- Inhalation of airborne radioactive particulates,
- Inhalation of aerosol indoor radon progeny (in the case of the presence of radon predecessors) and tritiated water vapor (the radon pathway was turned off by Rancho Seco because the NRC does not regulate dose received from radon and progeny),
- Inadvertent ingestion of radioactive material directly from the source, and
- Ingestion of materials deposited on the surfaces of the building compartments.

The containment building was modeled separately from the rest of the remaining structures. Rancho Seco evaluated two scenarios in determining the Building Surface DCGLs that would apply inside of the containment building as follows:

8.1.4 Building Renovation/Demolition Scenario

Because reasonable models of future containment building usage indicated very little occupancy, the building renovation scenario was modeled as being potentially bounding. The building renovation/demolition scenario as described in NUREG/CR-5512, Vol. 1 along with the input data template and input parameter values provided in ANL/EAD/03-1, specify the use of a volume source with a thickness of 15 cm. In the case of the containment building any residual contamination will likely be fixed on the interior surface rather than dispersed throughout the 15 cm thickness. If the assumption is made that containment building surface activity would be mixed into the 15 cm thickness during demolition, then DCGL values may be calculated by assuming that all of the activity contained in the source is actually on the surface. Using this methodology, the values in Table 8-1 were determined using the RESRAD-Build code. A large factor in this scenario is the occupancy time, which was reduced to by 63 days versus the standard value of 97.4 days for the unrestricted building occupancy scenario.

8.1.5 Limited Access Industrial Worker Scenario

Rancho Seco also analyzed an additional scenario, which considered that routine access to the inside of the containment building was extremely unlikely. The occupancy time in this scenario is based on the required time to inspect the building, which was assumed to be 4 days per year.

Table 8-1 lists the DCGLs determined for these two scenarios along with those using the standard assumptions for Building Occupancy without restrictions used at a number or plant sites. When comparing the results it can be seen that limiting the occupancy time increases the resulting DCGLs significantly. Although Rancho Seco could justify the Industrial Worker Scenario, for conservatism they applied the Building Renovation/Demolition Scenario to bound the possibility of that scenario occurring in the future. The use of scenario assumptions that place restrictions on the site, result in significantly higher DCGLs. This is expected to result in lower costs due to a facilitated Final Status Survey and less building remediation.

Table 8-1
Comparison of Rancho Seco Building Surface DCGL for Alternate Scenarios

Radionuclide	Renovation/Demolition Scenario DCGLs (dpm/100 cm^2) (See Note 2)	Limited Access Industrial Worker Scenario DCGL (dpm/100 cm^2) (See Note 2)	Industrial Worker Scenario DCGLs (dpm/100 cm^2)
H-3	1.21E+09	Note 1	3.15E+08
C-14	2.03E+08	Note 1	8.56E+06
Na-22	4.73E+04	Note 1	1.70E+04
Fe-55	6.25E+08	Note 1	3.42E+07
Ni-59	1.41E+09	Note 1	7.99E+07
Co-60	4.02E+04	8.90 E+05	1.52E+04
Ni-63	5.42E+08	Note 1	3.05E+07
Sr-90	2.01E+06	1.71 E+06	1.21E+05
Nb-94	6.60E+04	Note 1	2.29E+04
Tc-99	2.39E+08	Note 1	1.17E+07
Ag-108m	6.51E+04	Note 1	2.21E+04
Sb-125	2.63E+05	Note 1	7.99E+04
Cs-134	6.70E+04	1.05 E+06	2.19E+04
Cs-137	1.82E+05	2.29 E+06	5.56E+04
Pm-147	1.72E+08	Note 1	1.67E+07
Eu-152	9.19E+04	Note 1	3.18E+04
Eu-154	8.45E+04	Note 1	2.97E+04
Eu-155	4.38E+06	Note 1	5.23E+05
Np-237	1.71E+04	Note 1	2.38E+03
Pu-238	2.43E+04	8.06 E+04	3.42E+03
Pu-239	2.22E+04	7.29E+04	3.05E+03
Pu-240	2.22E+04	7.29E+04	3.05E+03
Pu-241	1.15E+06	3.77E+06	1.82E+05
Am-241	2.14E+04	7.08E+04	2.99E+03
Pu-242	2.31E+04	Note 1	3.20E+03
Cm-244	3.84E+04	Note 1	6.02E+03

Note 1: These Radionuclides were not included in the analysis.

Note 2: Applies only inside the containment structure.

8.1.6 Soil DCGLs

Under the Industrial Worker Scenario the average member of the critical group receives potential exposure from contaminated soil by direct exposure, inhalation of contaminated soil that becomes airborne and ingestion of contaminated soil. The industrial worker could also receive potential exposure from drinking water or buried piping.

The RESRAD code previously discussed was chosen as the computational method to calculate soil DCGLs. The Industrial Worker Scenario as used at Rancho Seco varies significantly from the Resident Farmer Scenario by allowing less conservative but realistic assumptions. Based in the Industrial Worker Scenario, the RESRAD pathways suppressed for Rancho Seco were:

- The plant ingestion pathway.
- The meat ingestion pathway.
- The aquatic foods pathway.

The drinking water pathway conservatively was not suppressed as one of the four potable water wells existing on the 2,480-acre site lies in the northern portion of the impacted area. Suppression of the drinking water pathway is not linked to the owner-control of the site.

RESRAD Version 6.3 (released in the summer 2005 by ANL) was selected by Rancho Seco to perform site-specific dose modeling of impacted area soils because of the ability to model subsurface soil contamination. Table 8-2 shows the DCGL calculated for the radionuclides determined to be significant for Rancho Seco. These radionuclides were determined based on highest analysis results for a soil sample taken at Rancho Seco. Table 8-2 also contains the corresponding Site Specific Soil DCGLs determined for CY and the NRC Generic Screening DCGLs. A comparison of the values shows considerable higher values for Rancho Seco. These results illustrate again how restrictions on the use of the site can increase numerical release limits.

Table 8-2
Comparison of Rancho Seco Soil DCGLs for Alternate Scenarios

Radionuclide	Rancho Seco Site Specific $DCGL_s$ (pCi/g) [1 pCi = 3.70E-2 Bq]	CY Site Specific Soil DCGL (pCi/g)	Generic Screening Soil DCGLs (pCi/g)
C-14	8.33E+06	5.66 E+00	1.2 E+01
Co-60	1.26E+01	3.81 E+00	3.8 E+00
Ni-63	1.52E+07	7.23 E+02	2.1 E+03
Sr-90	6.49E+03	1.55 E+00	1.7 E+00
Cs-134	2.24E+01	4.67 E+00	5.7 E+00
Cs-137	5.28E+01	7.91 E+00	1.1 E+01

8.1.7 Applicability of Surface Soil DCGLs to Sub-Surface Soil

Subsurface soil (i.e., soil at depths greater than 15 centimeters (5.9 in) below the soil surface) contamination was identified in limited areas within the Industrial Area at Rancho Seco. Therefore it was necessary for Rancho Seco to evaluate the applicability of the surface soil values to subsurface soil contamination. Rancho Seco determined that using surface soil DCGLs for large areas of contaminated sub-surface soil is slightly less-conservative. There is a 9.05 percent increase in calculated total dose by increasing the contaminated layer thickness from 0.15 meters to 0.5 meters. However, there is little additional increase in total dose by increasing the contaminated layer thickness up to 3 meters. This non-conservatism was discounted unless sub-surface soil contamination exists over a large area (greater than 300 m^2). At 300 m^2, the area factor for Cs-137 (the predominant dose contributor for the Table 4-12 radionuclide mixture) is 1.11, which is greater than the non-conservatism of 8.74 percent. The area factor for Cs-137 increases for areas less than 300 m^2 up to a factor of 11.3 for 1 m^2.

8.2 Buried Piping

The buried piping scenario used by Rancho Seco incorporates the soil DCGL values discussed above. Under this scenario, buried piping is assumed to disintegrate instantaneously upon license termination. The disintegrated media is assumed to be soil and the media volume is assumed to be equal to the piping volume. A gross DCGL value applicable to interior piping surfaces 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.

Potential dose to the receptor at one meter above the surface soil was evaluated assuming a soil cover depth of 0.305 meter and 1.0 meter. The latter depth is considered a nominal depth for buried piping that will remain on site after license termination. The MicroShield® computer code was used to perform these calculations. MicroShield® is a comprehensive photon/gamma ray shielding and dose assessment program.

8.3 Embedded Piping

Rancho Seco followed the accepted industry embedded piping scenario. 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 is 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 DCGLs are reduced as necessary by the dose contribution from the embedded piping in order to ensure compliance with the annual dose limit of 25 mrem/yr (0.25 mSv/yr.) The MicroShield® computer code was used to evaluate dose from embedded piping.

8.4 Bulk Materials

Although Rancho Seco considered the building surface DCGLs discussed above applicable to most structural surfaces, they considered the potential that some structural surfaces could contain volumetric contamination arising from neutron activation. They also considered the possibility that some volumetric contamination was caused by the migration of surface contamination into the materials of construction. Therefore, it was necessary to calculate DCGLs for bulk materials in order to evaluate these surfaces during the conduct of final status surveys.

Only portions of the RESRAD-Build dose model were considered by Rancho Seco to be appropriate for derivation of single nuclide DCGL values for activated or volumetrically contaminated bulk material. Because most interior concrete in the containment building, down to the carbon steel liner plate, was to be removed; only the carbon steel liner and concrete below it that are in the area formerly below the reactor vessel have a potential of being activated. Also, in other areas of the remaining structures, the floors would have the highest possibility of containing volumetric contamination due to spills of radioactive liquids. Therefore, only the floor area of 137 m^2 derived for the Building Surface DCGLs was used by replacing the floor surface source with a 1 foot thick (the most likely maximum depth of activation or contamination according to NUREG/CR-5884, Volume 2) volume source.

For the case of tritium in the volume sources, the tritium was assumed to be present in the volume sources in the form of water that is released from the volume sources in the form of vapor (HTO vapor). Table 8-3 lists the Bulk Material DCGLs for Rancho Seco.

Table 8-3
Rancho Seco Bulk Material DCGLs

Radionuclide	DCGL (pCi/g) [1 pCi = 3.70E-2 Bq]	Radionuclide	DCGL (pCi/g)
H-3	7.86 E+03	Cs-137	3.38 E+01
C-14	1.60 E+06	Pm-147	1.64 E+06
Na-22	8.39 E+00	Eu-152	1.64 E+01
Fe-55	3.91 E+07	Eu-154	1.50 E+01
Ni-59	1.49 E+07	Eu-155	7.81 E+02
Co-60	7.06 E+00	Np-237	7.49 E+01
Ni-63	6.85 E+06	Pu-238	3.61 E+02
Sr-90	4.16 E+03	Pu-239	1.23 E+02
Nb-94	1.18 E+01	Pu-240	2.96 E+02
Tc-99	7.37 E+05	Pu-241	2.05 E+04
Ag-108m	1.20 E+01	Am-241	2.70 E+02
Sb-125	4.75 E+01	Pu-242	3.09 E+02
Cs-134	1.22 E+01	Cm-244	6.72 E+02

8.5 License Condition to Rancho Seco Decommissioning Safety Analysis Report (DSAR)

The District has submitted 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 and related minimum detectable concentrations,
- 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.

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

8.6 Final Status Survey Operations

The conduct of the final status survey encountered certain challenges. The following will present details on some innovative uses of technology for the Final Status Survey (FSS.)

8.6.1 In-Situ Gamma Spectrometry

In-situ gamma spectrometry (In-situ) is an established technique for assaying the average radionuclide concentration in large volumes of material such as activated concrete. It has the advantage of being able to assess large areas with a single measurement. If desired, the detector's field of view can be reduced through collimation to allow assay of smaller areas.

In situ object counting refers to gamma spectrometry systems that include software capable of modeling photon transport in complex geometries for the purpose of estimating detector efficiencies. This eliminates the need for a calibration geometry representing the object to be counted. Such systems are also useful for assaying piping and complex components such as heat exchangers.

The use of in-situ methodology in place of scanning has been more widely used in Decommissioning Final Status Surveys as experience has grown and is used for many types of surveys at Rancho Seco. The following is an example of In-situ gamma spectroscopy being use in place of conventional scanning.

8.6.2 Rancho Seco Containment Dome Survey

One of more successful uses of ISOCS is the FSS of the inside of the Containment Dome at Rancho Seco. Previous experience in performing the FSS of the inside of the dome at the Trojan plant involved using conventional scanning performed from a large scaffolding structure that had been constructed on top of the polar crane. This challenge along with low DCGLs made a number of decommissioning plants decide to dispose of all the above grade portions of the containment dome as radioactive waste.

As discussed above, Rancho Seco's higher DCGLs (due to realistic scenarios associated with retained site ownership) helped to make ISOCS an acceptable alternative to scanning of the dome. Rancho Seco had decided against putting up such scaffolding in order to reduce personnel risk and to save project schedule time. The Rancho Seco approach allowed the survey of the liner to be performed with no loss of sensitivity and without having to place personnel at risk to build scaffolding upon the crane, climb the scaffolding to perform surveys, conduct remediation if necessary and then disassemble the scaffolding once the survey was complete. By employing *in-situ* gamma spectroscopy with a wireless-configured Multi Channel Analyzer (MCA), a small platform was placed on top of the polar crane which supported a remote-controlled man lift which could position the ISOCS detector at the necessary locations for performing the surveys of the dome interior surface without a technician riding the manlift (See Figure 8-1).

Figure 8-1
Final Status Survey of the Rancho Seco Containment Dome With ISOCS

The interior surface of the Rancho Seco Containment Building dome consists of the painted steel liner which extends from the spring line at elevation 115 feet (35 m) (approximately 20 feet [6 m] above the polar crane rail) to the top of the dome at elevation 158.5 feet (48.3 m.) The dome has a total surface area of approximately 1941 m^2. The liner had become contaminated during plant operation and had been subsequently decontaminated.

The survey design consisted of overlapping circular 28 m^2 measurements centered on the spring line of the dome and angling up one row at a time until the peak of the dome was reached. The survey was performed using a Canberra Industries, characterized, 40% relative efficiency HPGe detector. The detector geometry for the dome survey was defined as a 28 m^2 circular plane with a source to detector distance of 3 m.

The count times were set to achieve an MDC of approximately 2200 dpm/100 cm^2. Based on an assumed Cs-137 to Co-60 ratio of 80:20 (typical for other site structures) the dome individual nuclide DCGLs were 182,000 dpm/100 cm^2 for Cs-137 and 40,200 dpm/100 cm^2 for Co-60. The respective initial $DCGL_{EMC}$ (Elevated Measurement Concentration) values for Cs-137 and Co-60 were determined to be 2,712,000 and 554,000 dpm/100 cm^2 respectively given a 1 m^2 area and an area factor of 14.9 for Cs-137 and 13.8 for Co-60.

Using a technique previously employed at another decommissioning project, an Investigation Criterion was established for both Cs-137 and Co-60 such that a 1 m^2 elevated area located at the edge of the detector field of view would not go undetected. The Cs-137 to Co-60 ratio for the liner was initially estimated to be 80:20 based on the nuclide fractions determined for other structures on site. This ratio resulted in setting investigation levels of 102,000 dpm/100 cm^2 for Cs-137 and 20,000 dpm/100 cm^2 for Co-60.

The survey design showed that 110 overlapping circular plane measurements were needed to cover the entire dome surface from the spring line to the top of the dome. This resulted in a total surveyed surface area of 3113 m^2 for a dome interior surface area of 1941 m^2, which results in an overlap factor of 1.60 or 60%. At each measurement location at the spring line level, two smears were taken of the liner surface to determine the removable surface activity. Since the entire area was surveyed at minimum detectable concentration (MDC) values which were equivalent to those achieved using hand-held gas proportional instruments and provisions were made for detecting elevated areas, the survey was an acceptable Class 1 survey even though direct beta measurements were not taken for personnel safety reasons.

The only interference within the survey area was several brackets for the Emergency Upper Dome Air Circulator. The ductwork for the air circulator had been previously removed leaving only the welded brackets attached to the dome surface (several have been removed to facilitate demolition work). The surface area of each pair of brackets that face away from the detector is approximately 0.5 m^2 and the entire compliment is approximately 18 m^2. This surface area is less than 1% of the dome interior surface area. Most of the surface of the brackets is still within the overlapping detector fields of view and any small area not seen was considered insignificant.

The reactor building dome survey was conducted while reactor internals disassembly was occurring. This was necessary in order to determine at the earliest possible time whether further decontamination of the liner would be required. It also necessitated the use of the detector backshield and performing background subtraction on the spectra collected due the varying radiation levels in the reactor building caused by the internals work. The initial survey design called for a minimum of three background measurements to be taken at the first survey ring (i.e., springline). However, as the survey progressed, it became apparent that additional background measurements were needed. Some of the added backgrounds were subtracted from multiple measurements when it was apparent that large areas within a given survey ring had consistent background activity. Some background measurements were applied to individual measurements in areas with highly variable background.

8.6.3 Final Status Survey Results

MARSSIM states that *in situ* gamma spectroscopy may be used where gamma emitting radionuclides are present to demonstrate compliance with the release criterion. The NUREG also states "if the equipment and methodology used for scanning is capable of providing the same quality as direct measurements (e.g., detection limit, location of measurements, ability to record and document results), then scanning may be used in place of direct measurements.

The dome survey data showed that the residual activity on the liner surface following the decontamination in 2001 was less than the dome surface DCGL (mean Cs-137 activity of 12, 446 dpm/100 cm^2 (7% of the DCGL) and Co-60 activity of 5,927 dpm/100 cm^2 (15% of the DCGL)). When the data are unitized, the measurement fractions are all less than one. On average, the combined Cs-137 and Co-60 activity was less than the structure surface gross beta DCGL of 43,000 dpm/100 cm^2(Industrial Worker Scenario without limited access) and well below the surface DCGLs approved for inside containment that assumed restricted access as discussed above. None of the measurements exceeded the investigation criteria and therefore no significant elevated areas of liner activity were present. Removable surface activity averaged 468 dpm/100 cm^2 with a maximum value of 2193 dpm/100 cm^2, which clearly met the 10% criteria.

If the survey had been designed in the standard manner would have required 272 direct gross beta measurements covering a total area of 2.72 m^2. Rancho Seco concluded that the survey results met the requirements for a Class 1 survey and demonstrates that the residual activity on the upper portion of the liner met the release criterion of 25 mrem/yr (0.25 mSv/yr.)

9 CURRENT STATUS

As of the time of the filing of this report the decommissioning of the Rancho Seco Plant was in the final phase. The status of the decommissioning tasks as of the September 2007 was as follows:

- As previously discussed, the spent fuel and Greater Then Class C Waste had been placed in shippable canisters and were stored inside dry storage concrete modules at the ISFSI on the site.
- All underground piping that exceeded release limits has been removed. All remaining underground piping has been surveyed or determined to be non-impacted.
- All remaining embedded piping that have been surveyed has been decontaminated (if necessary.) At this time the embedded piping FSS has not been completed.
- All impacted systems (except underground and embedded) have been removed.
- Most all non-concrete structures on site have been removed.
- Room decontamination in the Auxiliary Building is in progress. Final Status Survey is performed as areas within the building are completed.
- Decontamination of the Spent Fuel Pool is in progress. The Final Status Survey will be performed when decontamination is complete.
- The demolition of all concrete in the Reactor Containment Building is in progress. The Final Status Survey will be performed when demolition is complete.
- The Interim Onsite Storage Building (IOS) is being prepared for long-term storage of Class B and C waste. All Class A waste is being prepared for shipment and the area surrounding the building is being prepared to be the remaining licensed area.
- The Reactor and Auxiliary Buildings will be sealed for the long term closure.
- Final Status Surveys (FSS) are being conducted for land area as the individual land area decommissioning activities are completed.

All decommissioning physical work is scheduled for completion by early 2008. It is expected that these FSS activities would be completed in the third quarter of 2008.

10 LESSONS LEARNED

The following are key lessons learned from the Rancho Seco Decommissioning Project:

- RSNGS Incremental Decommissioning approach was the result of lack of Decommissioning Funds. However, it proved to be very successful in terms of utilization of plant staff and cost savings. Additionally, the work proceeded from low contaminated SSC to highly active dismantling activities, which allowed the plant staff and support personnel to gain needed experience in the transition to full-scale decommissioning.
- Even with the low decommissioning fund, innovative thinking and the willingness to try new methods (such as the robotic AJW and mechanical tooling) coupled with an overall flexible project schedule led to a successful low-cost decommissioning project.
- The development of a formal "Hot Spot Removal Program" proved to be an effective tool in reducing the initial worker exposure related to decommissioning activities.
- RSNGS was late in implementing the use of a Nuclear Island for cooling of the spent fuel pool due to its waiting to move fuel to the ISFSI. However, even with this delay the use of the Nuclear Island concept proved to be cost effective and allowed other decommissioning activities to proceed.
- Reactor Internals Segmentation key experiences:
  - Rancho Seco used a wide selection of mechanical cutting equipment which provides an experience base for the use of a wider range of cutting tools in future decommissioning projects. All of the mechanical cutting equipment, other than the RMT, functioned acceptably with only minor debugging.
  - Testing of Reciprocating Machine Tool proved to be unreliable in terms assessing field cutting performance. This experience underscores the importance to duplicating actual field condition in the tool testing phase of the project.
- Low Level Waste Management Program was exemplary in terms of effective waste reduction techniques and overall disposal strategies. Examples of their approach can be seen in the segmentation of the large components (e.g. reactor head, steam generators and the reactor vessel).

- The innovative use of in-situ gamma spectroscopy (ISOCS) in assaying the containment dome resulted in significant cost savings compared to previous dome surveying projects. In previous projects significant resources (i.e. costs and time) and personnel risks were involved on the installation and use of major scaffolding to accomplish the task. Rancho Seco was able to avoid considerable costs and risks by employing an ISOCS with a wireless-configured Multi Channel Analyzer (MCA) attached to a remote-controlled manlift placed on top of the polar crane.
- SMUD's retention of the site property allowed their use of the Industrial Worker Scenario for license termination. This had a significant impact on their final status survey and license termination as it resulted in expectantly higher DCGLs than more conservative dose models such as the Resident Farmer Scenario.

11
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7. Federal Register, Volume 64, No. 234, dated December 7, 1999, Generic Radionuclide Screening Values.
8. Rancho Seco License Termination Plan, Rev. 0, April 12, 2006.
9. Code of Federal Regulations, Title 10, Part 20.1402, "Radiological Criteria for Unrestricted Use."
10. Code of Federal Regulations, Title 10, Part 50.82, "Termination of License."

A
RADIOACTIVE WASTE VOLUMES

A.1 Ranch Seco Lifetime Waste Quantities

Based on annual reports Rancho Seco has shipped 10,135 m^3 of waste (5,949 Curies) for disposal through 2006. During active decommissioning (since 1997) RSNGS has shipped 5,844 m^3
(522 Ci, 1.93E13 Bq). All decommissioning waste shipped has been Class A. There is currently in storage 31.6 m^3 (294 Ci. 1.09E13 Bq) of Class B waste and 61.0 m^3 (16,000 Ci, 5.92E14) of Class C waste. There is one container of GTCC waste stored at the ISFSI containing 10.7 m^3 (36,000 Ci, 1.33E15 Bq). The remaining decommissioning waste is estimated to be 11,400 m^3 of Class A waste (very low activity concrete debris).

Total Waste Disposal Volume Life of Plant through 2006	10,135 m^3
Total Decom. Disposal Waste (Class A) Volume through 2006	5,844 m^3
GTCC in Storage	10.7 m^3
Class C in Storage	61.0 m^3
Class B in Storage	31.6 m^3
Remaining Decom. Waste (Class A) for Disposal in 2007/8 (est.)	11,400 m^3

Note: A major fraction of the remaining decommissioning waste is tied to containment building concrete.

B
ESTIMATED DECOMMISSIONING AND FUEL STORAGE COSTS

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 B-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	

C
PROJECT TIMELINE

Key Event	**Date(s)**
Shutdown Announcement/Defueled Certification to NRC	August 1989
Decommissioning Plan Submitted	May 1991
NRC issued Decommissioning Order	March 1995
SMUD Board of Directors Approved Incremental	
Decommissioning	January 1997
PSDAR Submitted	March, 1997
Decommissioning Public Meeting	Fall 1998
Asbestos Insulation Removal	1994 to 2007
Subcontract to Decommissioning Support Contractor	April 1997 to July 08
SMUD Board of Directors Approved Continuing	
Decommissioning through 2008	July 1999
Transfer Fuel to ISFSI Pad	June 01 to Aug 02
Steam Generator/Pressurizer Removals	Mar 04 to Mar 05
Reactor Pressure Vessel Internal Segmentation Project	July 2004 to May 06
Reactor Pressure Vessel Cutup and Shipment	July 2006 to Jan 07
License Termination Plan Submitted	April 2006
License Termination Plan Approval	Nov 07 (Projected)
Final Status Survey/Physical Work Complete	June 08 (Projected)
NRC License Terminated for All Non-ISFSI Areas	Dec 08 (Projected)

Note: The Part 50 license will be reduced to include only the IOSB and about 1 acre surrounding it. The license termination will not occur until the B and C waste is shipped.

D
ALARA

Figure D-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. The total nuclear worker exposure during decommissioning is currently estimated to be less than 200 person-rem (2 person-Sv.) This estimate is significantly below the 1,215 person-rem (12.15 person-Sv) estimate of the FGEIS for immediate dismantlement and below the ten-year SAFSTOR estimate of 664 person-rem (6.64 person-Sv.)

These low doses are primarily the result of the decay time prior to beginning dismantlement and the short operating time (~6 EFPY). The ALARA organization continued to operate as if the plant was an operational plant, providing significant input to the dismantlement planning.

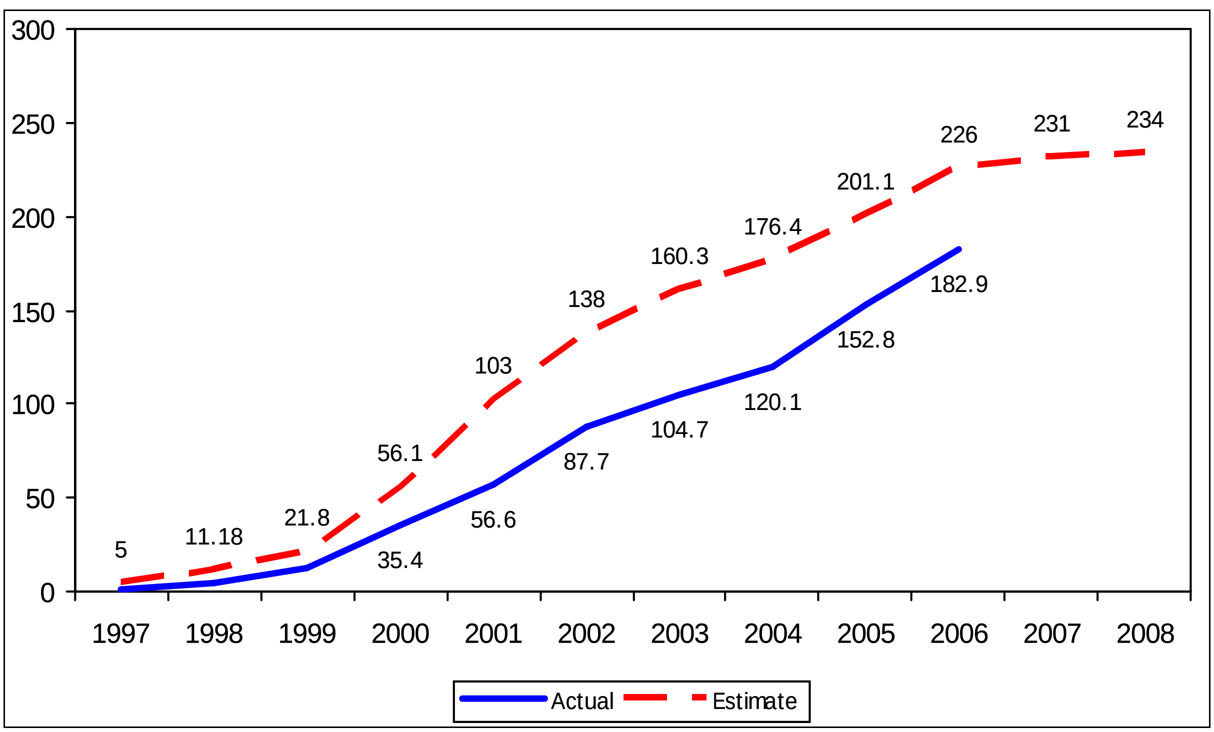

Figure D-1
Dose Estimates and Actual (Person-Rem)

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