ELECTRIC POWER RESEARCH INSTITUTE

Technical Bases for the Development and Application of Derived Concentration Guidance Levels for Decommissioning and License Termination of Nuclear Power Plants

2012 TECHNICAL REPORT

Technical Bases for the Development and Application of Derived Concentration Guidance Levels for Decommissioning and License Termination of Nuclear Power Plants

This document does NOT meet the requirements of 10CFR50 Appendix B, 10CFR Part 21, ANSI N45.2-1977 and/or the intent of ISO-9001 (1994).

EPRI Project Manager
K. Kim
Other EPRI Staff
Contributing to this Report
R. McGrath

ELECTRIC POWER RESEARCH INSTITUTE

3420 Hillview Avenue
Palo Alto, CA 94304-1338
USA

PO Box 10412
Palo Alto, CA 94303-0813
USA

800.313.3774
650.855.2121

askepri@epri.com

www.epri.com

1025315
Final Report, September 2012

DISCLAIMER OF WARRANTIES AND LIMITATION OF LIABILITIES

THIS DOCUMENT WAS PREPARED BY THE ORGANIZATION(S) NAMED BELOW AS AN ACCOUNT OF WORK SPONSORED OR COSPONSORED BY THE ELECTRIC POWER RESEARCH INSTITUTE, INC. (EPRI). NEITHER EPRI, ANY MEMBER OF EPRI, ANY COSPONSOR, THE ORGANIZATION(S) BELOW, NOR ANY PERSON ACTING ON BEHALF OF ANY OF THEM:

(A) MAKES ANY WARRANTY OR REPRESENTATION WHATSOEVER, EXPRESS OR IMPLIED, (I) WITH RESPECT TO THE USE OF ANY INFORMATION, APPARATUS, METHOD, PROCESS, OR SIMILAR ITEM DISCLOSED IN THIS DOCUMENT, INCLUDING MERCHANTABILITY AND FITNESS FOR A PARTICULAR PURPOSE, OR (II) THAT SUCH USE DOES NOT INFRINGE ON OR INTERFERE WITH PRIVATELY OWNED RIGHTS, INCLUDING ANY PARTY'S INTELLECTUAL PROPERTY, OR (III) THAT THIS DOCUMENT IS SUITABLE TO ANY PARTICULAR USER'S CIRCUMSTANCE; OR

(B) ASSUMES RESPONSIBILITY FOR ANY DAMAGES OR OTHER LIABILITY WHATSOEVER (INCLUDING ANY CONSEQUENTIAL DAMAGES, EVEN IF EPRI OR ANY EPRI REPRESENTATIVE HAS BEEN ADVISED OF THE POSSIBILITY OF SUCH DAMAGES) RESULTING FROM YOUR SELECTION OR USE OF THIS DOCUMENT OR ANY INFORMATION, APPARATUS, METHOD, PROCESS, OR SIMILAR ITEM DISCLOSED IN THIS DOCUMENT.

REFERENCE HEREIN TO ANY SPECIFIC COMMERCIAL PRODUCT, PROCESS, OR SERVICE BY ITS TRADE NAME, TRADEMARK, MANUFACTURER, OR OTHERWISE, DOES NOT NECESSARILY CONSTITUTE OR IMPLY ITS ENDORSEMENT, RECOMMENDATION, OR FAVORING BY EPRI.

THE FOLLOWING ORGANIZATION PREPARED THIS REPORT:

Electric Power Research Institute (EPRI)

THE TECHNICAL CONTENTS OF THIS DOCUMENT WERE **NOT** PREPARED IN ACCORDANCE WITH THE EPRI NUCLEAR QUALITY ASSURANCE PROGRAM MANUAL THAT FULFILLS THE REQUIREMENTS OF 10 CFR 50, APPENDIX B AND 10 CFR PART 21, ANSI N45.2-1977 AND/OR THE INTENT OF ISO-9001 (1994). USE OF THE CONTENTS OF THIS DOCUMENT IN NUCLEAR SAFETY OR NUCLEAR QUALITY APPLICATIONS REQUIRES ADDITIONAL ACTIONS BY USER PURSUANT TO THEIR INTERNAL PROCEDURES.

NOTE

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

Acknowledgments

The following organization prepared this report:

Electric Power Research Institute (EPRI)
3420 Hillview Ave.
Palo Alto, CA 94304

Principal Investigator
R. McGrath

This report describes research sponsored by EPRI.

The Electric Power Research Institute would like to acknowledge Empresa Nacional de Residuos Radiactivos (Enresa) for sharing experiences related to the ongoing decommissioning of the Jose Cabrera Nuclear Power Plant (JCNPP) Site. In particular the assistance of the following persons is greatly appreciated:

- Juan Luis Santiago – Projects Department Head
- Nieves Martín – JCNPP Decommissioning Project Manager
- Elisa Gimeno – Collaborator on Jose Cabrera D&D Project

This publication is a corporate document that should be cited in the literature in the following manner:

Technical Bases for the Development and Application of Derived Concentration Guidance Levels for Decommissioning and License Termination of Nuclear Power Plants. EPRI, Palo Alto, CA: 2012. 1025315.

Product Description

Nuclear Power plants achieve license termination by meeting regulatory site release criteria. Depending on the country, these criteria may be based on radionuclide concentration or dose. For dose-based criteria, corresponding radionuclide concentration limits, called Derived Concentration Guidance Levels (DCGLs), must also be developed. This report provides information related to site release criteria and the development of DCGLs.

Background

The ultimate goal of the decommissioning of a nuclear power plant site is to release the site for future use. The primary mechanism of that release is the removal of the power plant site from the control of the regulatory license. Regulations provide the criteria that the site must meet for release for future use. These criteria can be dose based or radionuclide concentration based and can be site-specific to the intended future use of the site. In the United States, the Nuclear Regulatory Commission's License Termination Rule requires that licensees decontaminate and remediate sites as needed so that future users of the site will not receive more than 25 millirem (or 0.25 millisieverts) per year of dose. European and Asian countries may follow Clearance Levels established by government agencies or develop country specific dose based regulations. For dose based site release criteria, site-specific radionuclide concentration limits are developed for any media to be left on-site (e.g. concrete, soils, groundwater, etc.)

Objectives

To investigate and provide experiences, lessons learned, and examples related to the development of site-specific DCGLs at decommissioning nuclear power plant sites.

Approach

The project team investigated experiences from decommissioning and decommissioned nuclear power plants in the United States and Spain. These power plants have completed decommissioning or are in the process of developing site-specific DCGLs for their sites.

Results

Several nuclear power plants in the U.S. have completed decommissioning and obtained license termination from the Nuclear Regulatory Commission (NRC) using the 25 millirem (0.25 millisievert) per year site release criteria. In the U.S., licensees may apply the NRC's generic DCGLs or develop site-specific DCGLs based on the reasonable, foreseeable future use of the site (e.g. Resident Farmer, Industrial, etc.) These site-specific DCGLs must be reviewed and approved by the NRC. The use of site-specific DCGLs can facilitate decommissioning while maintaining protection of a future user of the site. The Spanish nuclear regulatory body, Consejode Seguridad Nuclear (CSN), also allows the development and implementation of site-specific and future use-based DCGLs. The Jose Cabrera Nuclear Power Plant will be the first to develop and implement DCGLs in Spain, pending review and approval by CSN.

This report provides methods and technical bases behind the DCGLs implemented at various U.S. nuclear power plant sites (such as Big Rock Point, Trojan, Connecticut Yankee, Rancho Seco, etc.) and a Spanish nuclear power plant (Jose Cabrera). Some of these plants implemented generic DCGLs while others developed site-specific DCGLs. The report describes DCGLs developed for soil, groundwater, buildings, and bulk materials (such as concrete) and the dose models applied to the media. It also discusses site-specific variables that impact the development of DCGLs for each site.

Applications, Value, and Use

Licensees planning for or actively decommissioning nuclear power plant sites can apply the methods and technical bases provided in this report as guidance and benchmarking for developing and implementing DCGLs for their sites. DCGLs are not only applicable to decommissioning nuclear power plant sites but can also be useful tools and metrics for operating nuclear power plants evaluating soil and groundwater remediation options.

Keywords

Decommissioning
Remediation
Site release criteria
License termination
DCGL

Table of Contents

List of Figures

List of Tables

Section 1: Introduction

The ultimate goal of the decommissioning of a nuclear power plant site is to release the site for future use. The primary mechanism of that release is the removal of the power plant site from the control of the regulatory license. In the U.S., this release is primarily governed by the federal U.S. Nuclear Regulatory Commission (NRC) through the License Termination Rule (LTR) (Reference 1-1) which was implemented in 1996. This regulation instituted dose based site release criteria [more commonly known as the 25 mrem/yr (0.25 mSv/yr) criteria] for termination of an NRC license. Since that time, a number of NRC guidance documents have been issued showing acceptable compliance methods. Also over that time, considerable experience has been gained at decommissioning power plants in the development of radionuclide concentration based site release limits that correspond to the site release criteria and the conduct of the Final Status Surveys to meet those limits.

The goal of this report is to present the evolution of regulatory guidance and the experiences gained in developing radionuclide concentration based site release limits, otherwise known as the Derived Concentration Guidance Levels (DCGLs). This experience report will present approaches and case studies from actual U.S. nuclear power plant sites that have been decommissioned and proposed site release limits in Spain. Finally, this report describes simplified methods to develop draft site release limit that can be used by plants in the early phases of decommissioning or by operating plants to evaluate characterization data and remediation options at a site.

In addition to the detailed information concerning the development of soil DCGLs, this report contains a summary of information concerning the development of site release limits for other media such as structures, embedded piping, and buried piping. Additional information of site release limits and their application during Final Status Survey are contained in the following EPRI Reports:

- Final Status Survey Experience Report (EPRI Report 1015500, Reference 1-2)
- Concrete Characterization and Dose Modeling During Plant Decommissioning (EPRI Report 1015502, Reference 1-3)
- Characterization and Dose Modeling of Soil, Sediment and Bedrock During Plant Decommissioning (EPRI Report 1019228, Reference 1-4)
- Use of In-situ Gamma Spectroscopy During Nuclear Power Plant Decommissioning (EPRI Report 1021108, Reference 1-5)

Section 2: U.S. Nuclear Regulatory Commission Site Release Regulations and Guidance

2.1 Overview

In order to understand the site release limits that will apply at license termination, the regulations and regulatory guidance that apply to that release must be understood. In addition to federal regulations, there may also be other stakeholders (such as local or state regulators) that have input into the standards used to release a nuclear power plant from regulatory control. This section will provide a summary and references for these regulations, regulatory guidance, and other potential stakeholder implications.

2.2 Site Release Regulations

2.2.1 U.S. Nuclear Regulatory Commission Regulations

2.2.1.1 U.S. Nuclear Regulatory Commission Criteria for Unrestricted Release of a Site

The U.S. Nuclear Regulatory Commission (NRC) site release criteria (Reference 1-1) defines the standard that a site to be released for unrestricted use must meet. There is an additional requirement that an evaluation be performed to demonstrate that it does not meet the As Low As Reasonably Achievable (ALARA) criteria to remediate any areas of the site to a lower dose level. "Unrestricted use" means that there are no restrictions on the use of the site after the operating license is terminated. The criteria of this regulation requires that the residual radioactivity that is distinguishable from background radiation allowed to remain on a site at the time of license termination would result in a Total Effective Dose Equivalent (TEDE) to an average member of the critical group that does not exceed 25 mrem per year (0.25 mSv per year) from all dose pathways. Two definitions of the terms used in this regulation are:

- "Average Member of the Critical Group" is defined by the NRC in the 10 Code of Federal Regulations (CFR) 20.1003 (Reference 1-1) as "the group of individuals reasonably expected to receive the greatest exposure to residual radioactivity for any applicable set of circumstances." This term applies to the user of the site after it has been released from its NRC license. The amount of time this person is present on the site (i.e. occupancy rates) and his sources of food and water (i.e. consumption actions) affect the projected dose that the person will receive from the residual radioactivity left on the site after license termination. The anticipated future use of the site affects the occupancy on the site and whether any food or water from the site is consumed by the average member of the critical group. The average member of the critical group is normally assumed to be an adult male.

- "Total Effective Dose Equivalent (TEDE)" is the total dose to an individual, considering direct exposure, inhalation, and ingestion dose. (Reference 1-1) To determine the TEDE dose, the various types of dose received by different organs of the body are adjusted to the same basis as the whole body dose using weighting factors.

2.2.1.2 U.S. Nuclear Regulatory Commission Criteria for Restricted Release of a Site

The NRC regulations for restricted release of a site require that the licensee has made provisions for legally enforceable institutional controls that provide reasonable assurance that the TEDE from residual radioactivity distinguishable from background (to remain on site at the time of license termination) to the average member of the critical group will not exceed 25 mrem/year (0.25 mSv/year). Restricted release means that the site has in place, at the time of license termination, institutional controls that will limit the occupancy rates and consumption actions of users of the site after termination of the license. The institutional controls need to be legally enforceable and are generally enforced by a government agency. The control of occupancy rates and consumption actions for users of the site will generally result in higher radionuclide concentration site release limits (or Derived Concentration Guidance Levels, DCGLs). This is because the institutional controls eliminate or reduce certain exposure pathways and leads to less projected dose to the member of the critical group per unit of residual radioactivity.

2.2.2 Other Stakeholder Considerations

In addition to regulations and regulatory guidance, other stakeholders for a particular site may have an input into the site release criteria to be applied at the time of decommissioning. Later in this chapter, case studies are provided for plant sites in Connecticut, Maine and Massachusetts where input from state agencies and/or the Environmental Protection Agency (EPA) resulted in site release standards that were more stringent than those required by the NRC. More details on the effect of other stakeholders on site release limits are contained in other EPRI Reports (References 1-2, 1-3 and 1-4).

Some additional potential impacts on site release limits are:

- Site ownership - The owner of the site may also have requirements on the release of the site that are more stringent than the federal or state criteria. The situation at the San Onofre Nuclear Generating Station described in

Section 3.3.8 is an example of this. These additional requirements could result in lower DCGLs and more remediation being required for the site.

- Stakeholders relationships - Other stakeholders such as local or state government authorities or other organizations can also be given the ability to provide input into the determination of site release limits. This is another potential impact on the final DCGLs approved for a site.

2.2.3 Derived Concentration Guideline Levels (DCGLs)

- Per NRC guidance, the radionuclide concentrations that correspond to the NRC site release criteria of 25 mrem/yr (0.25 mSv/yr) are called Derived Concentration Guideline Levels, or DCGLs. DCGLs are the concentrations of radionuclides that can remain on-site in various media (i.e. groundwater, soil, buildings) after site release and license termination that meet the site release criteria for the site. DCGLs are the radionuclide concentrations that decommissioning nuclear power plants remediate to at the time of decommissioning.

As the dose to a future user of the site is too low to measure directly, the projected post closure dose from radionuclide concentrations is determined by numerical models based on the future use scenario of the site. These calculations are generally complex and carried out by computer codes. How these codes are used to determine the DCGLs is described later in this report in Sections 4.3.2 and 5.3.2 and 6.2.

2.2.4 Radionuclides of Interest

- To determine the numerical site release limits, the radionuclides of interest, or the radionuclides that may contribute to dose to the future user, at a site need to be defined. This list will also be used in determining the analytical requirements for samples collected for the purpose of assessing contamination on a site. The following are some approaches that can be used to determine the radionuclides of interest for a site:

- As part of a sites groundwater monitoring program, a list of radionuclides for analysis should have been determined. This list can also be used as the list of radionuclides of interest for determining site release limits. If this list has not been prepared for the site, NUREG-1757, “Consolidated Decommissioning Guidance, Volume 2” (Reference 2-1), specifically Appendix O, provides guidance for identifying a suite of radionuclides that could be present at a power reactor.

- The process discussed in NUREG-1757 is used to determine which radionuclides could be present at the site in concentrations that, if left un-remediated, would result in a dose to a future user of the site that is a significant fraction of the site's release limit. The number of radionuclides included in this list has generally been about 20 for sites being decommissioned. As an example, the list of radionuclides of interest at the Connecticut Yankee (CY) plant during its decommissioning is provided in Table 4-3. Once this list has been developed, single radionuclide soil site release limits can be chosen or determined through dose modeling.

It should be noted that the listing of radionuclides of interest in Table 4-3 does not include most of the short-lived radionuclides normally detected in samples at operating power plants. Past experience has shown that a considerable time (on the order of at least 5 to 10 years) will have passed from the time of final shutdown of the plant to license termination. During this time, the short-lived radionuclides prevalent during plant operations will have decayed to negligible concentrations.

2.3 U.S. Nuclear Regulatory Commission Guidance

The U.S. Nuclear Regulatory Commission (NRC) published regulatory guidance for compliance to the License Termination Rule (LTR, Reference 1-1) soon after the Rule was issued in 1996. The early guidance documents described a limited number of scenarios and parameters for use in dose modeling calculations. These scenarios and the resulting site release limits are described in Sections 4 and 5.

The NRC monitored experiences and lessons learned by plants as they implemented the revised dose modeling methods under the LTR. The NRC issued additional guidance based on these experiences and lessons learned supporting the use of realistic dose modeling scenarios which "...could result in more economical decommissioning while continuing to maintain safety."(NRC SECY-03-0069, Reference 2-3).

The following sections detail the revised NRC guidance, which was published in 2003 through 2006, concerning the use of realistic dose modeling scenarios.

2.3.1 Revised U.S. Nuclear Regulatory Commission Guidance for Realistic Future Land Use Scenarios

This NRC Policy Issue, SECY-03-0069 "*Results of the License Termination Rule Analysis*" (Reference 2-3) issued in 2003 describes the results of an analysis performed by the NRC concerning the experiences observed in the implementation of the License Termination Rule. In the NRC Policy Issue, the NRC staff identified one significant source of potential conservatism was seen as the difficulty in selecting and justifying land use scenarios for the 1000-year dose assessment time period. Previous guidance had stated that the selection and justification of land use scenarios were to be based on a 1000-year dose assessment time period. The uncertainty in such a long period is thought to have resulted in the selection of very conservative default scenarios, such as the Resident Farmer Scenario, by licensees. The NRC staff recommended an option that included identifying reasonably foreseeable land use scenarios that are likely within the foreseeable future (e.g., the next few decades and to possibly 100 years), considering advice from land use planners and stakeholders. This option would also identify less likely, alternate scenarios to the reasonably foreseeable scenarios, to understand the robustness of the analysis.

- In NRC Regulatory Issue Summary 2004-08, *Results of the License Termination Rule Analysis,* (RIS 2004-08, Reference 2-4) the NRC

Commissioners approved the recommendations of SECY 03-0069 (Reference 2-3). This allowed licensees to justify scenarios based on reasonably foreseeable future land use as opposed to defaulting to the very conservative scenarios.

The NRC based this revision on an NRC review of dose modeling experiences of the plants that had begun decommissioning prior to 2003. Larger power plant sites that began their decommissioning prior to 2003 include Trojan, Connecticut Yankee, and Maine Yankee. Discussions on how these plants chose their land use scenarios are provided in Chapter 3. Chapter 3 also discusses how the Rancho Seco site chose a land use scenario based on "realistic" exposure scenarios after issuance of this SECY-03-0069 guidance. The Big Rock Point site chose its land use scenario around the time that the NRC first issued its updated guidance on this topic in NRC Policy Issue, SECY-03-0069. Discussion of the Big Rock Point experience is included in Section 3.3.5.

2.3.2 NUREG 1757, "Consolidated Decommissioning Guidance"

The philosophies contained in SECY 03-0069 (References 2-3) and RIS 2004-08 (Reference 2-4) were incorporated into the 2006 revision of to NUREG 1757, "Consolidated Decommissioning Guidance" (Reference 2-1). NUREG 1757 provides detailed guidance on the conduct of decommissioning. The following summarizes NUREG 1757 concerning the establishment of exposure scenarios for use in dose modeling:

- The compliance criteria in 10 CFR Part 20 for decommissioning does not require an investigation of all (or many) possible scenarios
- The compliance scenario may be based on a bounding scenario, such as, a screening scenario or another scenario using conservative assumptions about land uses or behaviors, or be based on the reasonably foreseeable land uses for the area.

NUREG 1757 defines the time frame for "reasonable foreseeable land uses" as within 100 years. As stated in the SECY and RIS, the NUREG also states that less likely but plausible scenarios may be analyzed by the licensee, not for compliance to regulations, but to ensure a robust, risk-informed decision.

Section 3: Site Future Use Decision

3.1 Overview

The first step in developing site release limits for decommissioning a nuclear power plant site is to define what the future use of the site will be. Key factors that effect the selection of specific future use scenarios at decommissioning nuclear utilities are discussed in the following sections.

3.2 Site Release Limit Background

When the U.S. Nuclear Regulatory Commission (NRC) implemented the License Termination Rule (LTR) in 1996, there were limited options for dose models and site use scenarios described in NRC guidance. The only scenario for land use described extensively in this early NRC guidance was the Resident Farmer Scenario. With the revised guidance published in 2003 to 2006, more realistic scenarios such as the Industrial Worker Scenario were developed. The next to sections describe those scenarios.

3.2.1 Resident Farmer Scenario

The Resident Farmer Scenario assumes that a family resides at the site after license termination. This family obtains all of its food from the site (raises livestock for meat, catches fish, and grows vegetables) and utilizes a well drilled on site to obtain its drinking and irrigation water. Figure 3-1 shows an illustration of the various dose pathways for the Resident Farmer Scenario included in the RESRAD dose modeling code. To determine Derived Concentration Guidelines Levels (DCGLs), the NRC D&D code or the RESidual RADioactivity (RESRAD) dose modeling computer code developed by Argonne National Laboratory are generally used. The D&D code can only calculate dose from surface soil. The RESRAD code can calculate dose for all pathways that are the result of residual radioactivity in surface soil, subsurface soil and/or groundwater. Due to uncertainty in obtaining NRC approval of the model results, the very conservative, default input parameters of these codes were often used by utilities in the late 1990's and early 2000's. As will be discussed in Chapter 4, this approach resulted in very low DCGLs which, in some cases, required extensive remediation to achieve.

Information on the D&D code is readily available on the Oak Ridge Associated Universities website (Reference 3-1). Likewise information on the RESRAD family of dose modeling codes is contained on the Argonne National Laboratory website (Reference 3-2). Both codes can be downloaded and used free of charge from their respective websites.

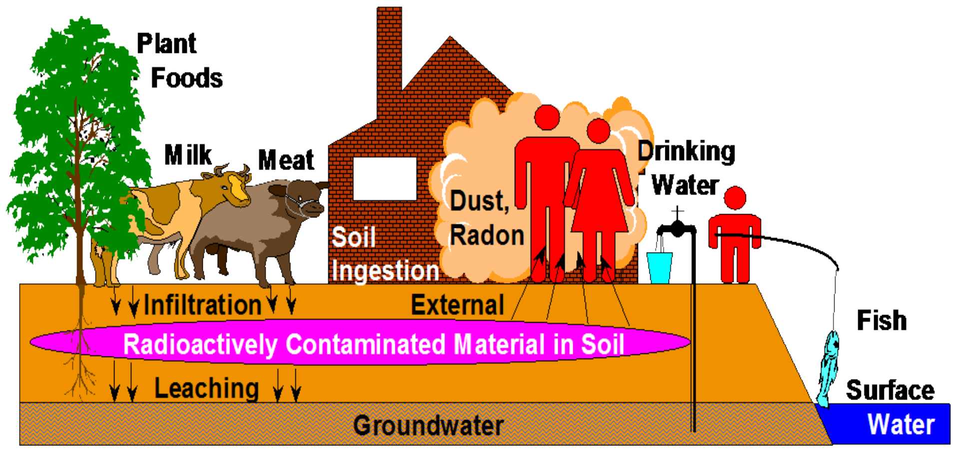

Figure 3-1
Dose Pathways of RESRAD Dose Modeling Code (Resident Farmer Scenario)

3.2.2 Industrial Worker Scenario

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. Figure 3-2 from NUREG-1757 "Consolidated Decommissioning Guidance" (Reference 2-1). depicts most of the typical exposure pathways for the Industrial Use Scenario.

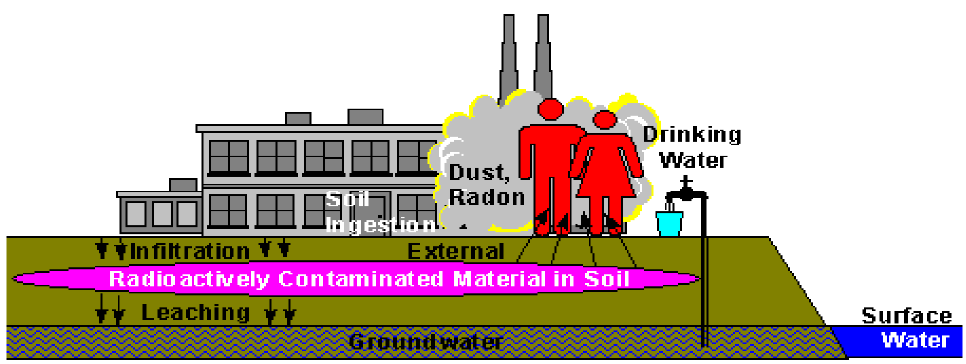

Figure 3-2
Dose Pathways of RESRAD Dose Modeling Code (Industrial Worker Scenario)

The Industrial Worker Scenario varies significantly from the Resident Farmer Scenario by allowing less conservative but realistic assumptions. Based on the Industrial Worker Scenario, the RESRAD pathways typically suppressed are:

- The plant ingestion pathway

- The meat ingestion pathway
- The milk ingestion pathway
- The aquatic foods pathway

3.3 Site Future Use Decision Case Studies

3.3.1 Trojan Nuclear Plant Site – Resident Farmer Scenario

The Trojan Nuclear Plant was a 1,130 MWe, four-loop Pressurized Water Reactor (PWR) located on a 623 acre (252 hectare) site in northwest Oregon on the Columbia River. This plant was operated by Portland General Electric Company (PGEC). As a result of an economic decision, which was driven in part by the need to replace the steam generators, the plant permanently shut down in 1993 after approximately 17 years of commercial operation.

The Trojan plant was the first to seek license termination under the NRC License Termination Rule (LTR) implemented in 1996. In fact, Trojan originally prepared a Decommissioning Plan (DP) and sought regulatory approval based on NRC release standards published in the pre-LTR guidance document, NRC Regulatory Guide 1.86, "Termination of Operating Licenses for Nuclear Reactors, June 1974" (Reference 3-3). Upon issuance of the LTR, Trojan needed to amend its DP to reflect dose based release limits instead of the fixed limits published in Regulatory Guide 1.86.

Although the utility had no plans to convey the site to another owner, they also had no plans for future use of the site other than for interim storage of the spent nuclear fuel in a dry fuel storage facility (i.e., Independent Spent Fuel Storage Installation, ISFSI). The decommissioning fund for the site was sufficient to allow the owners, PGEC, to proceed with prompt decommissioning.

Due to limited regulatory guidance available at the time, PGEC decided to utilize the generic radionuclide screening DCGLs published in the Federal Register as release limits (Reference 3-4 and discussed further in Chapter 4). The site performed a computer code dose modeling analysis using the NRC D&D code (See Chapter 4) with its default parameters to allow ease of review of the screening DCGLs by the NRC by showing the basis for screening DCGLs. The default parameters of the D&D code are based on the Resident Farmer Scenario.

3.3.2 Yankee Rowe - Resident Farmer Scenario

The Yankee Nuclear Power Station (commonly called Yankee Rowe) was owned and operated by Yankee Electric Power Company (YEPC) and was the only plant operated by that license holder. It was constructed from 1957 to 1960 and commenced power operation in late 1960. The plant was a 4 loop PWR of Westinghouse design. Its final output was 185 MWe.

Yankee Rowe was one of a number of plants in the northeastern portion of the United States owned by a consortium of New England utilities. Since the owners wanted to remove the liability of owning a nuclear power plant, they decided to promptly decommission the plant once it was permanently shut down in 1991. YEPC also wanted to convey all but the small area of the site associated with the ISFSI to another owner shortly after the release from the NRC license.

For the land area of the plant, Yankee Rowe utilized the Resident Farmer Scenario as its dose modeling scenario. This scenario was chosen, at least in part, so as not to limit or restrict the use of the site by potential future purchasers and owners. Choosing the conservative Resident Farmer Scenario would allow the site to be used for any uses that correspond to less restrictive exposure scenarios.

3.3.3 Maine Yankee – Resident Farmer Scenario

The Maine Yankee plant, owned and operated by the Maine Yankee Atomic Power Company was an 864 MWe, was a 3 loop PWR designed by Combustion Engineering and located on an 820 acre (332 hectare) site. It commenced power operation in 1972 and was permanently shut down in 1997. During its operating years, the plant was a significant electrical supply contributor to the New England electrical grid.

The Maine Yankee situation and future use decision was very similar to that at Yankee Rowe. The single unit site had many of the same owners as Yankee Rowe and was the only plant operated by the Maine Yankee Atomic Power Company. As with the Yankee Rowe site, the decision was made to promptly decommission Maine Yankee and release the site for unrestricted use (except for the ISFSI). The licensee planned to convey the majority of the site to a different owner as soon as possible. Maine Yankee assumed the Resident Farmer Scenario for dose modeling so as not to limit the use of the site by potential future purchasers of the site.

3.3.4 Connecticut Yankee

The Haddam Neck Plant, (Commonly called Connecticut Yankee or CY) was largely owned by the same consortium of New England electric utilities that owned Maine Yankee and Yankee Rowe. CY, a single unit facility, was located on a 525 acre (212 hectare) site in Haddam, Connecticut and housed a Westinghouse 4-loop PWR rated at 1,825 MWt and 619 MWe. This facility was operated by the Connecticut Yankee Atomic Power Company.

CY began commercial operation in January of 1968 and was permanently shut down in December of 1996. CY, having the same owners as the Yankee Rowe and Maine Yankee plants, made the same business decision as those two plants. In order to transfer most of the site areas to a new owner as soon as possible and not restrict the future use of the site, CY assumed the Resident Farmer Scenario for dose modeling.

3.3.5 Big Rock Point – Modified Resident Farmer Scenario

Big Rock Point was a 75 MWe Boiling Water Reactor (BWR) located on a 564 acre (228 hectare) site in Charlevoix, Michigan on the northern shore of Michigan's lower peninsula. The plant was operated by Consumers Energy from 1965 until it began decommissioning in 1997. The following information was taken from the Big Rock Point License Termination Plan (Reference 3-5).

The critical group for site-specific analysis of the Industrial Area at Big Rock Point was considered to be a modified resident farmer who moves onto the site. Some this resident farmer's diet consist of plants grown in a garden on the site. This resident farmer uses water tapped from the bedrock aquifer beneath the site. This resident farmer would not consume animal products raised onsite.

This Modified Resident Farmer Scenario was applied since the Big Rock Point Industrial Area was located in an area that is considered highly unlikely to ever be used for subsistence farming. The lakeshore of Little Traverse Bay in Lake Michigan is highly developed for summer residence and recreational uses. In addition, there were no Lake Michigan shoreline farms within 20 miles (33 km) of Charlevoix. Only 10.1 percent of Charlevoix County land was used for agricultural purposes and the county has an established declining trend in land use for agricultural purposes. Also, lakeshore soils in the area are poorly suited for subsistence farming because the soil is a gravelly-sandy loam containing low natural fertility and a moderately low organic content. Finally, lakeshore property values would effectively prohibit the use of the site for subsistence farming and it is likely that the future use of the site would be resort or recreational use.

Based on the above justification, Big Rock Point requested approval to suppress the meat and milk pathways in the RESRAD dose model. The NRC approved this request indicating a willingness to accept realistic land use and dose modeling scenarios.

3.3.6 Rancho Seco – Industrial Use Scenario

The Rancho Seco Nuclear Generating Station owned and operated by Sacramento Municipal Utility District (SMUD) on a 2,480 acre (1,004 hectare) site was a 913 MWe Babcock and Wilcox, B & W, designed 2-loop PWR that began commercial operation in 1975. It ceased operation in June of 1989 based on a county referendum and entered a SAFSTOR status to allow for the accumulation of decommissioning funds. SAFSTOR is an NRC defined decommissioning strategy where the plant is placed in a safe condition after permanent shutdown and active decommissioning is delayed until a later time. Following the successful efforts begun in 1997 to remove some steam systems, the SMUD Board of Directors authorized full decommissioning in July of 1999. At the completion of decommissioning, all plant equipment has been removed but most structures have been left in place. The spent fuel has all been placed in dry storage in an ISFSI located onsite.

The situation and the future use decision made for the Rancho Seco plant was very different from the other plants' situations discussed above. Rancho Seco was owned by a municipal utility and there were no plans to convey the property to another entity. Also, areas of the site have been reused for a natural gas fired power plant and other SMUD controlled facilities. For the above reasons, the future use of the Rancho Seco site was determined to be industrial use. SMUD has demonstrated this commitment by the construction of a photovoltaic generating facility (Figure 3-3) in addition to the natural gas fired power plant (Figure 3-4) on the site. As will be discussed in Section 4, these site use decisions allowed the Industrial Use Scenario to be used in determining the release limits for the Rancho Seco site.

Figure 3-3
Rancho Seco Photovoltaic Generating Facility

Figure 3-4
Rancho Seco Combine Cycle Power Plant

3.3.7 José Cabrera Nuclear Power Plant (Spain) - Industrial Use

The José Cabrera Nuclear Power Plant (also known as the Zorita Plant) is a single loop, 160 MWe, Westinghouse PWR. The plant is located on the River Tajo approximately 44 miles (70 km) east of Madrid, Spain. The plant began commercial operation in 1968 and was permanently shut down on April 30, 2006. The plant was operated by Gas Natural Fenosa (formally known as Union Fenosa Generation). After moving the spent fuel from the plant to dry storage in 2009-2010, Gas Natural Fenosa (GNF) transferred the plant license to Empresa Nacional de Residuos Radioactivos, Sa (ENRESA) for decommissioning. Upon completion of the decommissioning, the license will be returned to GNF. Certain projects were conducted by ENRESA under contract to GNF during the transition period between plant shutdown and transfer of the license to ENRESA in 2010. Included in those projects was the characterization of radionuclide contamination on the site and the development of preliminary site release limits as will be described in the following.

In Spain, the radiological release for a site is set by the Spanish regulator, Consejode Seguridad Nuclear (CSN). Spanish regulations require that licensees evaluate two scenarios for developing site releases limits for site land areas: 1) the scenario for the intended future use of the site and 2) a scenario where the intended future use of the site is not followed. The dose limits for buildings and these two site land area scenarios are provided as follows:

- Buildings are to be remediated to the RP 113 Clearance Levels (Reference 3-6) and therefore no dose modeling is necessary. These Clearance Levels correspond to an effective dose of 10 uSv/yr (1 mrem/yr)
- Intended Use of the Site: For site land areas, the total dose from surface soil, subsurface soil, surface water and groundwater can be no more than 0.1 mSv/yr (10 mrem/yr) when the intended future use of the site is assumed (i.e., for example, the intended future use of the site is industrial use for José Cabrera)
- Intended Use of the Site Not Followed: For site land areas, the total dose can be no more than 1 mSv/yr (100 mrem/yr) if the intended future use of the site is not followed and a scenario resulting in higher dose occurs (i.e., for example, a scenario that would result if the institutional controls for industrial use at José Cabrera fail).
- The utility owner of the José Cabrera site (Gas Natural Fenosa, GNF) has committed to an industrial use of the site after the decommissioning has been completed. As will be discussed in Section 4.3.2.1.3, this has allowed the dose modeling for soil at the site to be based on the Industrial Worker Scenario. As described in Section 4.3.2.1.3, José Cabrera assumed that the alternative scenario, "loss of institutional control" (which is postulated to occur at the time of license termination), results in the site being used by a Resident Farmer. The dose to the Resident Farmer from the "Industrial Use" soil concentrations was calculated so as to make sure that the dose due to this loss of industrial control would be no more than 1 mSv/yr (100 mrem/yr).
- In order to account for the potential for dose from groundwater, nuclide specific Groundwater DCGLs were developed following the approach used at Connecticut Yankee (see Section 4.3.2.1.1).

3.3.8 San Onofre Nuclear Generating Station, Unit 1 – Future Use Scenario To Be Determined

The San Onofre Nuclear Generating Station, Unit 1 (SONGS 1) was a 410 MWe Westinghouse PWR owned (80%) and operated by Southern California Edison Company (SCE). San Diego Gas & Electric Company (SDG&E) owns the remaining 20% of the SONGS 1 facility. The unit began commercial operation in 1968 and was permanently shutdown in November 1992. SONGS 1 shared a site with two other operating nuclear units. Major decommissioning activities began in 1999 when it was determined that space on the site was needed for an Independent Spent Fuel Storage Facility (ISFSI) for all three SONGS units.

The situation and decision for the SONGS 1 plant presents another variation on the end use decision. The property on which SONGS is located is owned by the United States military. Under a lease agreement with the military, after the site has been closed and released from its license, it is to be returned with essentially all structures removed (it has already been negotiated that the discharge piping will remain). Another difference from other decommissioned plants to date is the presence of two operating nuclear power plants on the site. As these plants are expected to operate at least another 20 years, the site will not seek release from any of the NRC licensees until all the remaining plants are shut down. Other shutdown plants in the U.S. that have been in this situation (i.e. of a shutdown unit on a site with operating units) have generally chosen to place the shutdown plant in a SAFSTOR condition and plan on decommissioning them along with the other units. The need for space for the ISFSI did not allow SONGS to take this approach and prompt decommissioning of Unit 1 was conducted instead.

3.4 Summary of Site Future Use Decision Case Studies

This chapter has discussed the different approaches used at power plant sites being decommissioned to decide the intended future use of the sites after termination of the NRC license. Among the approaches used are Resident Farmer Scenario, Modified Resident Farmer Scenario, and Industrial Use Scenario. Chapters 4 and 5 will discuss how these decisions and other factors such as the interests of stakeholders other than the NRC have affected the development of site release limits.

Section 4: Dose Modeling to Determine Site Release Limits for Land Areas

4.1 Overview

This chapter will discuss the general basis used at decommissioning nuclear power plants in the United States for selecting dose modeling scenarios for land areas. How the U.S. experiences are being applied in Spain will also be discussed. These scenarios are then used to develop radionuclide concentration site release limits that meet dose based site release criteria. Also discussed are dose modeling experiences at nuclear power plant sites since the implementation of the License Termination Rule in 1996.

4.2 Options for Development of Land Area Site Release Limits

Once the decision on the future use of the site has been made per the discussion in Chapter 3, radionuclide concentration limits that meet the site release criteria can be determined. As the dose to a future user of the site is too low to measure directly, the projected post closure dose from radionuclide concentrations is determined by numerical models. These calculations are generally carried out by computer codes.

A plant site has a number or options in determining the site release limits to be used during different phases of the decommissioning planning and implementation. Some of the options are discussed in the following sections.

4.2.1 NRC Published Screening Values for Soil

The NRC has published screening values for soil that have been calculated using conservative default dose modeling input parameters in NUREG 1757, "Consolidated Decommissioning Guidance" Volume 2 Revision 1, Table H.2 (Reference 2-1). The soil screening values for most of the radionuclides that have been found to be significant during past decommissioning projects are listed in Table 4-2. These screening values were determined using the NRC DandD dose modeling code, which was developed by the Sandia National Laboratories (information on the DandD code is contained in Reference 3-1). The default parameters of the DandD code are based on the Resident Farmer Scenario (described in Section 3.2) as the projected future use of the site. By using the default input parameters and the conservative Resident Farmer Scenario, the screening values are considered pre-approved by the NRC for use by a site during the decommissioning process. Less conservative DCGLs can generally be calculated by a site using site-specific input parameters and more realistic future site use scenarios (for example, by using the RESRAD code as described above in section 3.2.)

In NUREG 1757 (Reference 2-1), the NRC staff is cautioned to verify that the following conditions exist for each of the residual contamination conditions before approving the use of the screening values at a site. As such, these are also good guidance for application by the licensee:

- The initial residual radioactivity (after decommissioning) is contained in the top layer of the surface soil [e.g., approximately 15 centimeters (6 inches)].
- The unsaturated zone and the groundwater are initially free of residual radioactivity.
- The vertical saturated hydraulic conductivity at the specific site is greater than the infiltration rate (e.g., there is no ponding or surface run-off).

To address these potential limitations on the use of the NRC screening values, plant sites have provided justifications or adjusted the screening values to obtain NRC approval. The following provides a summary of some of these case studies that are examples of justifications that could be used or have been accepted by the NRC:

- Contamination is only surface soil – The Rancho Seco power plant reviewed the affect of contamination existing to the subsurface deeper than the 6 inch (15 cm) depth (considered surface soil during its decommissioning). Rancho Seco calculated that the surface soil DCGLs for their site were approximately 9% non-conservative when the contaminated area was large (greater than 300 m^2 [360 yd^2]) and assumed to be 10 feet (3 meters) thick. When the size of the contaminated area was limited to 300 m^2 (360 yd^2), the area factor (the allowable multiplier of the DCGL for areas smaller than the 10,000 m^2 [default assumed size of the survey unit] used in RESRAD to calculate the DCGLs) for Cs-137 (the predominant dose contributor for the radionuclides of interest at Rancho Seco) is 1.11. As long as the contaminated area is not larger than 300 m^2, the 9.05% non-conservatism from subsurface soil is more than compensated for by 11% higher allowable DCGLs for the 300 m^2 area. 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 (10 ft^2). As these possible increases in dose from subsurface soil contamination are a relatively small fraction of the total and only exist for large contaminated areas, not including the effect of increasing contaminated soil thickness in the development of the DCGLs at Rancho Seco was approved by the NRC. Although this example was for DCGLs determined using the RESRAD code, this justification could also be used for use of the NRC Screening Values for subsurface soil.
- Groundwater is initially free of contamination – a number of plant sites including Connecticut Yankee, Maine Yankee and Yankee Rowe received

NRC approval to compensate for the existence of groundwater contamination at the time of the Final Status Survey at the site by reducing the dose allowed from soil contamination by the projected dose from groundwater. The details of the methodology used are presented in Section 4.2.2. Of particular note is the case at Maine Yankee where the site utilized the NRC published soil screening values for both surface and "Deep Soil" (subsurface) soil. The general approach was to set the allowable soil concentration after dose from all the other media had subtracted from the 10 mrem/yr (0.1 mSv/yr) release criteria (this lower dose criterion was required by the State of Maine). Once the dose allotted for soil was determined, the NRC published screening values were multiplied by the ratio of the soil dose limit chosen and 25 mrem/yr (0.25 mSv/yr) to obtain the adjusted values. This same approach is described in Section 4.2.1.1 to illustrate how to adjust the NRC Screening Values for the effect of other media such as groundwater contamination.

- The vertical saturated hydraulic conductivity at the specific site is greater than the infiltration rate (e.g., there is no ponding or surface run-off) – To address this limitation, the sites being decommissioned that have contaminated areas that exhibit this condition have not used the NRC published screening values in those areas or have provided additional justification that the NRC screening values are conservative for those situations.

It should be noted that NRC screening values (and all DCGLs) are for individual radionuclides and that when mixtures are present; the "Sum of Fractions" rule must be applied. It should also be noted that DCGLs are not adjusted due to impacts from other site areas. Each site is divided into a number of survey units. Each survey unit must meet the DCGLs approved for the site.

The NRC also noted in NUREG-1757 (Reference 2-1) that the use of the single default parameter set for all radionuclides could result in overly conservative limits. The user is instructed in the NUREG that tailoring the default parameter set to individual radionuclides would, in most cases, result in higher DCGLs. Industry experience has shown this effect to be particularly evident for the transuranic radionuclides. This can be seen in Table 4-12 where the NRC soil screening values for the transuranic radionuclides are lower by a factor of approximately 10 compared to those determined by site specific analysis with RESRAD.

4.2.1.1 Adjusting NRC Screening Values for Potentially Contaminated Groundwater

As discussed in the previous section, the NRC Screening Values do not inherently account for groundwater contamination. As such, to be used for groundwater remediation purposes or as site release limits for sites with groundwater contamination, the NRC screening values must be adjusted. This section discusses an approach to the development of site release limits based on the NRC Screening values that have been adjusted to compensate for the contamination in ground water.

The first step in adjusting the NRC Screening Values is determining the dose resulting from the contamination in groundwater. The experience at decommissioning plants has typically been to establish the U.S. Environmental Protection Agency (EPA) Maximum Contamination Levels (MCLs) (Reference 4-5) as limits for radionuclide contamination in groundwater. For all the sites that have achieved release from their NRC licenses to date, the EPA MCL concentrations have not been exceeded either for all monitoring wells at the site or for the average of the groundwater concentrations within the capture zone of a drinking water well of a theoretical future resident of the site.

Table 4-1
Determination of Dose Due to Groundwater at the MCL Concentrations

Radio-nuclide	Maximum Contaminant Levels in Groundwater pCi/L (Bq/L)	CY 25 mrem/yr (0.25 mSv/yr) TEDE Groundwater DCGLs pCi/L (Bq/L) (Section 4.2.2.1.1)	TEDE Dose Due to MCL Concentrations in Groundwater (Calculated using CY Groundwater DCGLS) mrem/yr (mSv/yr)
H-3	20,000 (740)	6.52E+05 (2.4E+04)	0.77 (0.0077)
C-14	2,000 (74)	9.01E+03 (3.3E+02)	5.55 (0.0555)
Mn-54	300 (11)	2.42E+04 (9.0E+02)	0.31 (0.0031)
Fe-55	2,000 (74)	6.54E+04 (2.4E+03)	0.76 (0.0076)
Co-60	100 (3.7)	1.14E+03 (4.2E+01)	2.19 (0.0219)
Ni-63	50 (1.85)	3.15E+04 (1.2E+03)	0.04 (0.0004)
Sr-90	8 (0.30)	2.51E+02 (9.3E+00)	0.80 (0.008)
Tc-99	900 (33)	2.64E+04 (9.8E+02)	0.85 (0.0085)
Cs-134	80 (3.0)	3.42E+02 (1.3E+01)	5.84 (0.0584)
Cs-137	200 (7.4)	4.31E+02 (1.6E+01)	11.6 (0.116)
Eu-152	200 (7.4)	7.33E+03 (2.1E+02)	0.68 (0.0068)
Eu-154	60 (2.2)	5.05E+03 (1.9E+02)	0.30 (0.003)

Table 4-1 shows the EPA MCL concentrations for radionuclides of interest normally identified during power plant decommissioning. The Connecticut (CY) Groundwater Derived Concentration Guidance Levels (DCGLs), derived using site-specific dose modeling and calculations, are provided to show the radionuclide concentrations in groundwater that would result in a dose of 25 mrem/yr (0.25 mSv/yr). The last column of Table 4-1 shows the calculated Total Effective Dose Equivalent (TEDE) dose that corresponds to the MCL concentrations for each radionuclide calculated using the CY Groundwater DCGLs (see Section 4.2.2.1.1 for more information on the CY Groundwater DCGLs) as follows:

MCL TEDE Dose (mrem/yr) = (MCL Concentration x 25 mrem/year) / CY Groundwater DCGL *Eq. 4-1*

A review of Table 4-1 shows that only 4 radionuclides (C-14, Co-60, Cs-134 and Cs-137) result in a TEDE dose greater than 1 mrem/yr (0.01mSv/yr) at the MCL concentrations. Of these four radionuclides only Cs-137 has been present in significant concentrations in groundwater when a power plant was approved for license termination. At Connecticut Yankee, the highest monitoring well exhibited a groundwater Cs-137 concentration of 29.8 pCi/L (1.1 Bq/l) or 15% of the MCL just prior to license termination. When the effect of all radionuclides detected in this worst case well at CY was included, the total TEDE dose was 2.4 mrem/year (0.024 mSv/yr).

The other three radionuclides (C-14, Co-60, and Cs-134) that have a TEDE dose at the MCL concentration that exceeds 1 mrem/year (0.01 mSv/yr) have not been detected in significant concentrations in groundwater due to the following:

- C-14 and Cs-134 were only detected in very low concentrations in soil and would therefore not be expected to be present in detectable quantities in groundwater
- Co-60 has been detected in elevated quantities in soil but has not migrated into groundwater in significant quantities due to its relatively high affinity for soil (i.e., high K_d)
- When the experience at all plants, including operating plants, is considered, the following generalizations can be made:
- If contamination is detected in groundwater at a power plant, H-3 is detected in essentially all cases.
- For a few sites, in addition to H-3, Sr-90 has been detected in concentrations higher than its MCL
- Detection of Cs-137 in groundwater at CY was an unusual situation due to the relatively low affinity of the backfill sand at CY for the radionuclide (K_d determined to be 16.8).
- All other radionuclides listed in Table 4-1 have not been detected in groundwater at power plants in concentrations higher than a small fraction of the MCL if detected.

Each licensee should make an assessment of the potential or actual radionuclides present in the groundwater at their site at the time of decommissioning and license termination.

Once the potential or actual radionuclides in groundwater and their associated limits have been determined, the soil site release limits are adjusted to reflect the radionuclides in groundwater. For example, considering that the most likely detected radionuclides at a power plant are H-3 and Sr-90, a conservative bounding approach for adjusting the NRC Screening Values for soil to account for the effect of groundwater contamination is the following:

- The TEDE dose at the MCL concentrations for both H-3 and Sr-90 is approximately 0.8 mrem/yr (0.008 mSv/yr) (and the "sum of the fractions" rule applies to showing compliance with the MCLs). Reduce the NRC soil screening values to reflect a projected dose of 2 mrem/year (0.02 mSv/yr) from groundwater contamination at the time of decommissioning. The additional 1.2 mrem/year (0.012 mSv/yr) of allotted dose for radionuclides in groundwater is to account for any potential radionuclides other than H-3 and Sr-90 .
- An additional reduction of 2 mrem/year (0.02 mSv/yr) in the dose allotted to soil [for a total reduction of 4 mrem/year (0.04 mSv/yr)] should be made to account for the potential of radionuclides leaching out of concrete basements. This adjustment was required during the Connecticut Yankee, Maine Yankee and Yankee Rowe decommissioning. The value of 2 mrem/year (0.02 mSv/yr) is intended to bound the highest value from the three plants listed above (dose at CY was calculated as 1.58 mrem/year (0.0158 mSv/yr).
- If as a part of the groundwater monitoring program at a site, concentrations of radionuclides other than H-3 and Sr-90 were unexpectedly detected at large fractions of the MCL concentrations, the adequacy of the allowed dose from groundwater should be reevaluated.

As an example, Table 4-2 shows a listing of the NRC published screening values for radionuclides normal detected during a power plant decommissioning (based on a dose of 25 mrem/year [0.25 mSv/yr]), and in the third column, a listing of the screening values adjusted down by 16% to account for the dose allotted for contaminated groundwater and buried concrete. A site could also elect to a different allotment for the projected dose from groundwater and buried concrete based on the characterization data for the site and adjust the NRC screening values accordingly.

Table 4-2
Adjustment of NRC Screening Values for Groundwater/Buried Concrete Dose Allotment

Radio-nuclide	NRC Screening Values (25 mrem/yr (0.25 mSv/yr) - Resident Farmer) pCi/g (Bq/g)	Screening Values Adjusted for Dose from Groundwater & Buried Concrete (21 mrem/yr (0.21 mSv/yr) - Resident Farmer) pCi/g (Bq/g)	Radio-nuclide	NRC Screening Values (25 mrem/yr (0.25 mSv/yr) - Resident Farmer) pCi/g (Bq/g)	Screening Values Adjusted for Dose from Groundwater & Buried Concrete (21 mrem/yr (0.21 mSv/yr) - Resident Farmer) pCi/g (Bq/g)
H-3	1.1 E+02 (4.1)	9.2 E+01 (3.4)	Cs-134	5.7 E+00 (0.21)	4.8 E+00 (0.18)
C-14	1.2 E+01 (0.44)	1.0 E+01 (0.37)	Cs-137	1.1 E+01 (0.41)	9.2 E+00 (0.34)
Mn-54	1.5 E+01 (0.55)	1.3 E+01 (0.47)	Eu-152	8.7 E+00 (0.32)	7.3 E+00 (0.27)
Fe-55	1.0 E+04 (369)	8.4 E+03 (311)	Eu-154	8.8 E+00 (0.33)	7.4 E+00 (0.27)
Co-60	3.8 E+00 (0.14)	3.2 E+00 (0.12)	Pu-238	2.5 E+00 (0.09)	2.1 E+00 (0.08)
Ni-63	2.1 E+03 (77)	1.8 E+03 (65)	Pu-239	2.3 E+00 (0.08)	1.9 E+00 (0.07)
Sr-90	1.7 E+00 (0.06)	1.4 E+00 (0.05)	Pu-241	7.2 E+01 (2.6)	6.0 E+01 (2.2)
Nb-94	5.8 E+00 (0.21)	4.9 E+00 (0.18)	Am-241	2.1 E+00 (0.08)	1.8 E+00 (0.07)
Tc-99	1.9 E+01 (0.70)	1.6 E+01 (0.59)	Cm-243	3.2 E+00 (0.12)	2.7 E+00 (0.10)

4.2.1.2 Comparison of D&D and RESRAD Dose Modeling Codes

EPRI has published two reports concerning the use of the D&D and RESRAD dose modeling codes discussed above: "Use of Probabilistic Methods in Nuclear Power Plant Decommissioning Dose Analysis" (EPRI Report 1006949, References 4-1) and "Comparison of Decommissioning Dose Modeling Codes for Nuclear Plant Use: RESRAD and D&D" (EPRI Report TR-112874, Reference 4-2). These reports provide significant details how to run the codes and which input parameters have the greatest effect on the results. Reference 4-2 (specifically Section 6.1 and 6.2) provides 25 mrem/yr (0.25 mSv/yr) soil DCGLs using the RESRAD default input parameters. As the RESRAD default input parameters for contaminated zone thickness and depth to groundwater are 2 meters (6.6 feet) and 4 meters (13 feet) respectively, these results could be useful for a site having subsurface contamination and a fairly shallow groundwater depth. The soil DCGLs using the RESRAD default parameters have been included in Table 4-12.

Although there are other methods used to determine site specific Derived Concentration Guidance Levels (DCGLs), the RESRAD Dose Modeling Computer Code has been used by most utilities submitting site release limits to the NRC for approval during decommissioning. For this reason, the following discussion will focus primarily on the use of RESRAD in developing site specific soil DCGLs. NRC guidance on the use of computer codes to calculate DCGLs is contained in NUREG 1757, Volume 2, Revision 1, Chapter 5 (Reference 2-1). Section 4.2.2 includes examples of the use of RESRAD to calculate DCGLs at actual power plant site decommissioning.

RESRAD utilizes a number of input parameters to calculate site release limits. For many of these parameters, the code's default input parameter values can be used to determine estimated results. Varying these parameter values does not significantly affect the calculation result. However, for some parameter values, it is important to have site-specific inputs. For determining site specific DCGLs, site-specific values for the following parameter will likely need to be known or estimated as they have the potential to have a significant effect on the results:

- Soil Distribution Coefficients (K_d) for detected radionuclides. (This input parameter is more important if relatively mobile radionuclides such as Sr-90 are present in soil contamination and generally not important if only relatively immobile radionuclides such as Co-60 are present.)
- Depth to groundwater below the contaminated area
- Size (volumetric extent) of the contaminated area
- Hydrology in the contaminated area (e.g., soil porosity, hydraulic conductivity, hydraulic gradient, yearly precipitation)
- Assumed location of drinking water well (if applicable)

- Some of this information may have been collected by the site's groundwater monitoring program. If not, soil sampling (for K_d determination) and research of site characteristics may be needed.
- Table 4-12 provides a comparison of the NRC Soil Screening Values to examples of site specific soil DCGLs determined using the RESRAD Dose Modeling Computer Code. For the cases shown in columns 3 and 5 of Table 4-12, the affects of subsurface contamination was included. Under certain exposure scenarios such as these two for the Resident Farmer Scenario, the NRC Soil Screening Values are reasonably close to those that have been determined by site specific analyses. In these types of cases, the use of the NRC Screening Values may not result in significantly lower DCGLs for the site. Each site needs to evaluate its' soil characterization data before choosing to use the NRC Screening Values.

4.2.2 Site-Specific Dose Modeling Experience

- This section provides the reader with case studies concerning the development of site specific soil and groundwater Derived Concentration Guidance Levels (DCGLs) for sites that have been decommissioned. A site may wish to develop site-specific DCGLs for reasons such as the following:
- Certain characteristics of the site make the NRC screening values very conservative for that site (i.e., very deep groundwater table beneath the site, soil with relatively high K_ds that make radionuclides present in the soil relatively immobile.)
- Long term future use of the site is known or expected to be industrial, which will likely make certain dose pathways included in the development of the NRC screening values not applicable.

4.2.2.1 Dose Modeling for Soil

As discussed in Chapter 2 above, the NRC recommended the use of realistic exposure scenarios that reflected expected use of the site for the foreseeable future. The dose pathways to be analyzed are then set based on a realistic future use of the site. In order to illustrate actual plant experience before and after the updates to the NRC guidance, two actual cases for power plant sites being decommissioned will now be presented. The case involving Connecticut Yankee (CY) will show the experience for a complex site performing dose modeling calculation prior to the update to NRC guidance issued in 2003-2004. The Rancho Seco plant experience shows the use of the realistic scenario approach discussed in the updated NRC guidance.

Case studies from the Big Rock Point, José Cabrera (Spain), Maine Yankee, Saxton and Yankee Rowe plant decommissionings are also included in order to provide alternate approaches.

4.2.2.1.1 Connecticut Yankee Experience

Due to the presence of significant groundwater contamination at Connecticut Yankee (CY), dose from contamination in as many as three different media needed to be accounted for in certain land areas. These media were soil, groundwater and subsurface concrete. Most of the information in this section concerning CY was obtained from The CY License Termination Plan (Reference 4-6). Using a process similar to that discussed in Section 2.2.4, CY determined the listing of radionuclides of interest shown in Table 4-3.

The Table 4-3 list does not include most of the short-lived radionuclides normally detected in samples at operating power plants as these would decay to insignificant levels during the 5 to 10 years needed to complete a power plant decommissioning.

Table 4-3
Radionuclides of Interest for Connecticut Yankee

Radionuclide	Half-life (years)	Radionuclide	Half-life (years)
H-3	**12.33**	Cs-134	2.06
C-14	**5,730**	Cs-137	30.17
Mn-54	0.86	Eu-152	13.3
Fe-55	**2.69**	Eu-154	8.59
Co-60	5.27	Eu-155	4.96
Ni-63	**100**	**Pu-238**	**87.74**
Sr-90	**28.8**	**Pu-239/240**	**2.41×10^4**
Nb-94	2.03×10^4	**Pu-241**	**14.4**
Tc-99	**2.14×10^5**	Am-241	432.2
Ag-108m	1.27×10^2	**Cm 243/244**	**28.5**

Note: Bold indicates "hard to detect" radionuclides.

CY, at the time the License Termination Plan (LTP) was submitted in 2000, had not yet fully determined the level of groundwater contamination present or the concentrations projected after remediation efforts. In order to account for the three media contributing to dose and allow for flexibility in setting concentration targets for remediation, a "compliance equation" was established for use in the Final Site Survey (FSS) of land areas as follows:

$$\textit{Dose Total} = \textit{Dose Soil} + \textit{Dose Existing Groundwater} + \textit{Dose Future Groundwater} \qquad \text{Eq. 4-2}$$

Definitions of the terms of this equation are as follows:

- Dose Total: The combined TEDE dose to the average member of the critical group due to residual radioactivity above background from all dose pathways.

- Dose Soil: The portion of the dose from all pathways that is contributed by the soil related pathways.
- Dose Existing Groundwater: The portion of the dose from all pathways from residual radioactivity currently in groundwater on site that would still be present at the time of release of the site.
- Dose Future Groundwater: The portion of the dose from all pathways due to residual radioactivity that is projected to leach from the concrete buildings at CY and be present in groundwater at the time of site release. A discussion of the development of this dose calculation methodology is provided later in this Section.

It should be noted that some of the components of Equation 4-2 may not apply at all power plant sites or all locations on a site. For example the "Existing Groundwater" dose component may not apply in scenarios where the use of the site was controlled to preclude the use of groundwater or the groundwater was of poor quality that precludes consumption. The "Future Groundwater" component may not apply if no contaminated concrete in contact with groundwater was left on site at license termination. The "Future Groundwater" component may also not apply if groundwater sampling has been conducted for a sufficient period of time after concrete remediation to detect the leaching of radionuclides from the concrete into the groundwater. In such a case, the detected groundwater contamination from the concrete would be included in the "Existing Groundwater" dose component.

In order to meet this compliance equation to satisfy the NRC and the State of Connecticut site release criteria, the following methodology was used at CY:

- DCGLs corresponding to 25 mrem/yr (0.25 mSv/yr) were determined for each of the media as discussed later in this report.
- Target levels were set for each type of media by dividing up the total dose for the site release criteria of 19 mrem/yr (0.19 mSv/yr) TEDE [Note: The State of Connecticut required this more stringent site criterion (versus the NRC criteria of 25 mrem/yr (0.25 mSv/yr). For sites using the NRC criteria, the same approach could be used but setting the targets based on a total dose of 25 mrem/yr (0.25 mSv/yr)]. As the final status survey for the land areas that were impacted by "existing" and "future" groundwater dose needed to be conducted beginning in May of 2006, these target levels were set based on the projections of what the actual doses would be at the time of site release (expected to be in mid-2007). Sufficient post remediation (the important remediation had taken place from 2004 to 2005) groundwater and concrete sample data had been collected to allow the confident setting of the dose targets for the various media as follows:
    - The "existing groundwater" dose target was set at 2 mrem/yr (0.02 mSv/yr) TEDE using the Groundwater DCGLs for the areas affected by groundwater contamination. This dose was set based on groundwater monitoring sample results and to bound the groundwater concentrations expected to exist at the time of release of the site projected to be mid-2007.

- The target dose due to concrete media (i.e. "future groundwater" dose) was set at 2 mrem/yr (0.02 mSv/yr) based on the sampling that had been conducted on concrete to remain on site [it has been previously noted that the final calculated "future groundwater" dose at CY was 1.58 mrem/year (0.0158 mSv/yr)]. The target dose was set somewhat higher than the final calculated dose to allow other Final Status Surveys (FSSs) to be conducted in the areas affected by multiple media. In order to be able to conduct these FSSs, an adjusted soil DCGL was needed.

- Reducing the State of Connecticut criteria of 19 mrem/yr (0.19 mSv/yr) by the existing and future groundwater dose (2 mrem/yr [0.02 mSv/yr] each) left 15 mrem/yr (0.15 mSv/yr) for dose due to the soil contamination in areas affected by groundwater contamination and where contaminated concrete was present. The corresponding 15 mrem/yr Soil DCGLs were calculated by scaling from the Soil DCGLs for 25 mrem/yr (0.25 mSv/yr). These reduced "Operational Soil DCGLs" were then used as limits in the Final Status Survey of the applicable land areas. Dose allowed for soil was higher for areas where existing and/or contaminated subsurface concrete were not present.

The following provides summary descriptions on various dose scenarios used to calculate release limits for land areas at CY.

Dose Assessment Model – Soil

Connecticut Yankee utilized the Resident Farmer Scenario for modeling the dose pathways due to residual radioactivity in soil as discussed in Chapter 3. Due to the presence of subsurface and groundwater contamination at CY, the RESRAD code was used to determine the Soil DCGLs. To assure that only dose due to soil was included in this calculation, it was assumed that there was no contamination in groundwater at the time of release of the site from the NRC license (time=0 years) in the model calculation. RESRAD version 6.1, Probabilistic Version[1] was used to perform a parameter sensitivity analysis. Once the parameters that had a significant impact on the dose calculation were determined, a conservative value from the range given for the parameter in NRC guidance (Attachment C to NUREG/CR-6697, Reference 4-7) was used as an input to RESRAD version 5.91 (Deterministic Version). For parameters shown to have insignificant impact to the resulting calculated dose, median values from the parameter range were used in agreement with NRC guidance at the time (Note that more recent NRC guidance is to use the parameter range for parameters shown to have insignificant impact by the probabilistic sensitivity analysis).

Dose Assessment Model – Groundwater

As discussed above, the DCGLs for soil at CY were determined with the assumption that no groundwater contamination existed In order to account for groundwater contamination present at the time of license termination (as CY

---

[1] Note that later versions of the RESRAD codes are available on the Argonne National Laboratory website

planned to convey the property to a new owner soon after the release of site areas from the license), separate "Groundwater" DCGLs were determined. As with the Soil DCGLs above, the Argonne Lab's RESRAD dose modeling code was used to determine TEDE dose at an arbitrary pre-selected value of groundwater contamination for each of the 20 radionuclides of interest for CY.

The RESRAD code is typically used to calculate radiation doses (and DCGLs) for a source above the water table. Per the guidance of Argonne National Laboratory, to develop a dose model to determine the dose from residual radioactivity in groundwater, it is necessary to set certain RESRAD input parameters as follows:

- Time since placement of material = 1 year
- Time for calculations = 1 year
- Model for water transport parameters = Mass Balance (MB) model
- Distribution coefficient in the saturated zone = 0 cm^3/g

By doing so, the groundwater (well water) concentrations calculated by RESRAD will be found to be greater than or equal to the groundwater concentrations in equilibrium with the contaminated zone, under saturated conditions, and the time to the peak of the mean dose will be 0 years.

The equilibrium groundwater concentration associated with the contaminated zone was calculated using the principals of linear sorption theory described in Appendix H of the "Users Manual for RESRAD Version 6.0", (Reference 4-7) from which the following equation was derived:

$$c = \frac{1000 S_o \rho_b}{\left[1 + \left(K_d \rho_b / n\right)\right] n} \qquad \textit{Eq. 4-3}$$

where,

C = Equilibrium groundwater concentration (pCi/l)
S_o = Initial principal radionuclide concentration in contaminated zone (pCi/g)
ρ_b = Bulk density of contaminated zone (g/cm^3)
K_d = Distribution Coefficient of contaminated zone (cm^3/g)
n = Total porosity of contaminated zone (Fraction)

Also an arbitrary initial radionuclide concentration of 1 pCi/g (0.037 Bq/g) was used for the soil comprising the contaminated zone. This value does not affect the results of the calculation.

Groundwater DCGL Determination

To determine the Groundwater DCGLs at CY, the dose from all groundwater related pathways was determined for an assumed groundwater concentration of 10 pCi/Liter (0.37 Bq/L) and then scaled to determine the concentration that corresponded to 25 mrem/yr (0.25 mSv/yr). To determine the dose to be included in the compliance equation (Equation 4-2) the individual DCGL concentrations that corresponded to 25 mrem/yr (0.25 mSv/yr) were scaled to the groundwater monitoring well sample results. The resulting Groundwater DCGL values are shown in Table 4-4.

To determine the dose from the "existing groundwater" at CY, the monitoring well concentration which was within 100 meters (328 feet) of a survey area whose sample concentrations resulted in the highest calculated dose was used to show compliance with the NRC release criteria for that survey unit. The 100 meter distance was determined to be the maximum capture zone radius for a postulated water supply well drilled by a future resident farmer for the CY site.

Table 4-4
Connecticut Yankee Groundwater DCGLs [25 mrem/year (0.25 mSv/yr)]

Radionuclide	Groundwater DCGL pCi/l (Bq/L)	Radionuclide	Groundwater DCGL pCi/l (Bq/L)
H-3	6.52E+05 (2.4E+04)	Cs-134	3.42E+02 (1.3E+01)
C-14	9.01E+03 (3.3E+02)	Cs-137	4.31E+02 (1.6E+01)
Mn-54	2.42E+04 (9.0E+02)	Eu-152	7.33E+03 (2.1E+02)
Fe-55	6.54E+04 (2.4E+03)	Eu-154	5.05E+03 (1.9E+02)
Co-60	1.14E+03 (4.2E+01)	Eu-155	3.25E+04 (1.2E+03)
Ni-63	3.15E+04 (1.2E+03)	Pu-238	1.51E+01 (5.6E-01)
Sr-90	2.51E+02 (9.3E+00)	Pu-239	1.36E+01 (5.0E-01)
Nb-94	6.75E+03 (2.5E+02)	Pu-241	4.60E+02 (1.7E+01)
Tc-99	2.64E+04 (9.8E+02)	Am-241	1.32E+01 (5.0E-01)
Ag-108m	4.24E+03 (1.2E+04)	Cm-243	1.94E+01 (7.2E-01)

4.2.2.1.2 Rancho Seco Experience

As discussed in Section 3.3.6, the Sacramento Municipal Utility District (SMUD) does not plan to transfer the Rancho Seco Nuclear Generating Station to another entity. This condition allowed the average member of the critical group at Rancho Seco to be 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 under this "industrial worker scenario" 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 applied to evaluating exposure to contaminated surface soils and subsurface soils. Most of the information in this section was obtained from the Rancho Seco License Termination Plan (Reference 4-9).

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 on the Industrial Worker Scenario, the RESRAD pathways suppressed for Rancho Seco were:

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

It should be noted that Rancho Seco did not detect any radionuclide contamination in the groundwater monitoring wells on site. Therefore, Rancho Seco did not need to including existing groundwater concentrations in the calculation of soil DCGLs or determine separate Groundwater DCGLs as was done at Connecticut Yankee.

Rancho Seco used RESRAD Version 6.3 (released in the summer 2005 by Argonne National Laboratory) to perform site-specific dose modeling of impacted area soils because of the code's ability to model subsurface soil contamination.

Sensitivity Analysis of Detectable Radionuclides

A sensitivity analysis was performed first to identify the RESRAD input parameters that are sensitive in the Industrial Worker Scenario for the radionuclides that were detected in significant quantities in the highest concentration soil sample taken on the Rancho Seco site. These significant radionuclides for Rancho Seco are listed in Table 4-12 along with the method used to determine significance. This parameter selection process starts with the evaluation of specific RESRAD parameters. The selected parameter is then classified as behavioral, metabolic or physical. Guidance is contained in the RESRAD User Manual (Reference A-8) as to which parameters are in which category(s). Some parameters may belong to more than one of these classifications.

Physical parameters are determined by the source, its location, and geological characteristics of the site (i.e., these parameters are source- and site-specific). These include the hydrogeological, geochemical, and meteorological characteristics of the site. The characteristics of atmospheric and biospheric transport up to, but not including, uptake by, or exposure of, the dose receptor, would also be considered physical input parameters.

A behavioral parameter is any parameter whose value would depend on the receptor's behavior and the scenario definition. For the same group of receptors, a parameter value could change if the scenario changed (e.g., parameters for recreational use could be different from those for residential use).

If a parameter represents the metabolic characteristics of the potential receptor and is independent of the scenario chosen, it is classified as a metabolic parameter. The parameter values may be different in different population age groups. According to the recommendations of the International Commission on Radiological Protection, Report 43 (ICRP-43) (Reference 4-10), parameters representing metabolic characteristics are defined by average values for the general population. These values are not expected to be modified for a site-specific analysis because the parameter values would not depend on site conditions.

If the parameters were classified as behavioral or metabolic (and could be dependent on site conditions), site-specific parameter values were used if available. If the site specific values were not available, the default values contained in RESRAD v6.22 were used by Rancho Seco for performing sensitivity analyses.

If the parameters were classified as physical, then they were reviewed to determine if measured, site-specific values or look-up values based on soil type for the parameters are available.

If measured, look-up, or site-specific values for physical parameters were not available the parameters were then ranked by priority as 1, 2, or 3 where 1 represents high priority, 2 represents medium priority and 3 represents low priority. This ranking was the second step in the procedure used by Argonne to develop the probabilistic code. The parameter ranking has been documented in Attachment B to NUREG/CR-6697(Reference 4-11).

If the physical parameters were ranked as priority 3, the assigned default values in RESRAD v6.22 were used by Rancho Seco for performing sensitivity analyses. Two exceptions to this assignment exist for the parameters of field capacity and inhalation rate for which statistical parameter distributions were developed by Argonne.

Argonne has developed statistical parameter distributions for the physical parameters ranked as priority 1 or 2 (and for the two priority 3 parameters listed in the last sentence). These parameter distributions have been documented in Attachment C to NUREG/CR-6697 (Reference 4-11).

Once the parameter values and the statistical parameter distributions were loaded into RESRAD v6.22, the code was run in the probabilistic mode for the radionuclides of interest to identify the sensitive parameters. The absolute value of the partial ranked correlation coefficient (PRCC) of the peak of the mean dose (calculated by RESRAD and displayed in the code output) was then used to classify the parameters with statistical distributions as sensitive or non-sensitive. PRCC was chosen because NUREG/CR-6692 (Reference 4-12) recommends that it be used when nonlinear relationships, widely disparate scales, or long tails are present in the inputs and outputs. If the absolute value of the PRCC was greater than 0.25, then the parameter was classified as sensitive. If the absolute value of the PRCC was equal to or less than 0.25, then the parameter was classified as non-sensitive.

Finally, values for use in dose modeling for the physical parameters which have been determined to be sensitive by the above process, were selected following the guidance of NUREG/CR-6676 (Reference 4-13) as follows:

- If the PRCC was negative, the parameter to dose correlation is negative and the parameter value at the 25% quartile of the parameter distribution was selected as using a value lower than the mean value is conservative.
- If the PRCC value was positive, the parameter to dose correlation is positive and the parameter value at the 75% quartile of the parameter distribution was selected as using a value higher than the mean value is conservative.

Once all the parameters for dose modeling have been selected by the above process, these parameters are input into RESRAD and the code run to calculate the Derived Concentration Guidance Levels (DCGLs).

Calculation of Surface Soil DCGLs

Soil samples at Rancho Seco had shown only low levels of soil contamination. Only 6 radionuclides from the list of 26 radionuclides determined to be potentially present were detected in the most highly contaminated, pre-remediation soil sample at the site. Furthermore, Rancho Seco determined that the total potential dose from the 20 undetected radionuclides at Minimum Detectable Concentrations (MDCs) of the sample analysis was below 2.5 mr/yr (0.025 mSv/yr). Per NRC guidance, radionuclides can be ignored if the total potential dose from these ignored radionuclides is below 10 % of the total allowed dose (i.e., 25 mrem/yr (0.25 mSv/yr). Considering this, the development of nuclide specific DCGLs for these 20 radionuclides was unnecessary.

Single nuclide DCGL values are calculated by performing a RESRAD calculation for the 6 radionuclides detected in the Rancho Seco soil sample. The site-specific RESRAD v6.22 dose model was first loaded with parameters determined in the sensitivity analysis. For the remaining parameters (those determined to be non-sensitive by the process discussed in the last paragraph), the statistical parameter distributions from Attachment C to NUREG/CR-6697 (Reference 4-11) were used. RESRAD was then run in the probabilistic mode for the radionuclides. The uncertainty analysis input settings for these calculations were:

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

These calculations provided the peak of the mean dose in mrem/year per pCi/g for each of the radionuclides detected in the highest activity soil sample at Rancho Seco. DCGL values for each detected radionuclide were then calculated by dividing the allowable dose (i.e., 25 mrem/yr) by the calculated peak of the mean dose (in mrem/year per pCi/g) to yield the DCGLs for each radionuclide.

Applicability of Surface Soil DCGLs to Sub-Surface Soil

As discussed in Section 4.3.1, Rancho Seco was able to show that the method used to determine the Rancho Seco Surface Soil DCGLs envelopes the effects of subsurface contamination that is up to 3 meters (10 feet) thick as long as the contaminated area is no larger than 300 meters^2 (360 yard^2). As the increases to dose due to the subsurface contamination are a relatively small fraction of the total, not including the effect of increasing contaminated soil thickness in the development of the DCGLs at Rancho Seco was approved by the NRC.

Alternative Exposure Scenario - Resident Farmer

Although it is considered highly likely that SMUD would continue to maintain the Rancho Seco site for industrial purposes indefinitely, the impact of releasing the site under the industrial worker scenario and subsequently allowing a member of the public to establish a subsistence farm on the site was evaluated by Rancho Seco.

The Resident Farmer exposure scenario for Rancho Seco calculated potential dose from the same exposure pathways used in the Industrial Worker exposure scenario:

- Direct exposure to external radiation from the contaminated soil material;
- Internal dose from inhalation of airborne radionuclides; and
- Internal dose from ingestion of:
    - Drinking water from a contaminated well, and
    - Contaminated soil.

In addition to potential dose calculated from the above exposure pathways, the Resident Farmer exposure scenario also calculates potential dose from the additional exposure pathways:

- Internal dose from ingestion of:
    - Plant foods grown in the contaminated soil and irrigated with contaminated water,
    - Meat and milk from livestock fed with contaminated fodder and water, and
    - Fish from a contaminated pond.

A sensitivity analysis was performed for the Resident Farmer scenario in the same manner as for the Industrial Worker scenario as described earlier in this Section with the additional dose pathways included.

Calculated Dose from Detected Radionuclides for a Resident Farmer Scenario

Once the sensitive parameters were identified for the Resident Farmer scenario, the parameter sensitivity model was revised. The sensitive parameter statistical distributions were replaced with the deterministically assigned sensitive parameter values. Then the model was run to calculate dose under the Resident Farmer scenario using the Industrial Worker scenario maximum allowable radionuclide soil concentrations. The dose was calculated for 0, 1, 3, 25, 50, 75, 100, 300, 500 and 1000 years after license termination and site release. This calculation assumed uniform contamination over the entire impacted area of the site. The results of the dose calculations are plotted on Figure 4-1. Figure 4-1 shows that the dose from the maximum allowable concentrations allowed by the Industrial Worker Scenario will be 25 mrem/yr (0.25 mSv/yr) approximately 30 years after the projected release of the site when using the Resident Farmer Scenario.

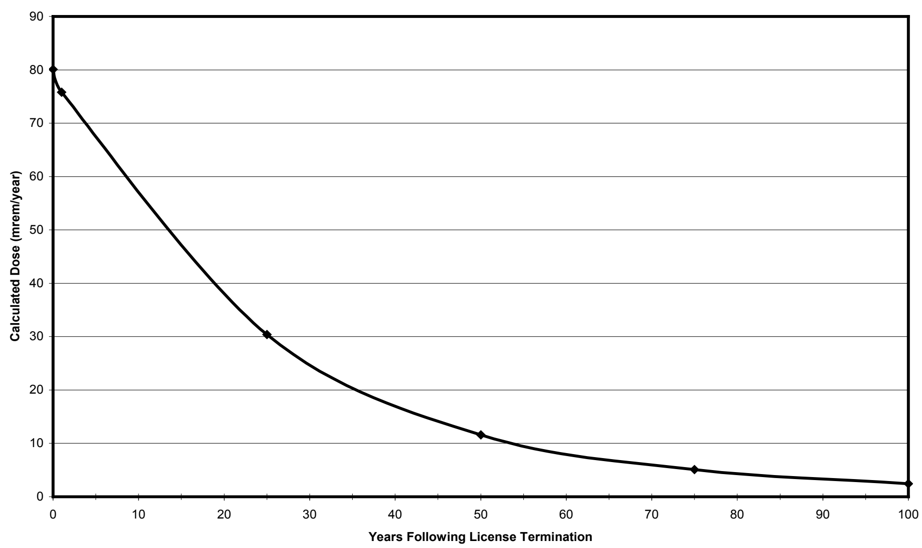

Figure 4-1
Dose for Resident Farmer Scenario Using Rancho Seco Industrial Worker Soil DCGLs

It is highly unlikely that SMUD would consider transfer of all or any portion of the impacted area of the site to another entity immediately upon the release of all but the Interim Spent Fuel Storage Installation (ISFSI) from the NRC license. At the time the DCGL analysis was performed, this partial release of the site was targeted to occur on July 1, 2008. Thirty years following the release of non-ISFSI areas of Rancho Seco from the NRC license was considered to be a reasonable time period during which the District would not transfer all or any portion of the impacted area of the site to another entity. This was reasonable since SMUD had made capital investment in new construction on the site. If the site was to be transferred to another entity 30 years after the release (i.e. July 1, 2038) and a subsistence farm was established on the site, the dose to this Resident Farmer would not exceed 25 mrem/yr (0.25 mSv/yr).

The reduction in dose from residual radioactivity over time illustrated by Figure 4-1 is noteworthy for operating plants and permanently shutdown sites that are in a SAFSTOR mode (i.e., to be decommissioned at a later date). It shows how a combination of radioactive decay and the redistribution of contamination due to precipitation and groundwater flow can reduce the dose effect from a certain concentration of radionuclides in soil. It should be noted that Figure 4-1 is based on the contamination being in surface soil and the hydrogeological conditions at Rancho Seco (i.e., relatively low quantity of precipitation and relatively deep groundwater table). To determine this effect at another plant, the actual location of the contamination and the conditions for that site should be used to determine the site specific redistribution effect.

4.2.2.1.3 José Cabrera Plant Experience

The initial approach that ENRESA used to calculated soil DCGLs for the José Cabrera Nuclear Power Plant site was similar to that used at Rancho Seco and Connecticut Yankee. ENRESA utilized an Industrial Worker Scenario as was used at Rancho Seco and developed Groundwater DCGLs as was done at Connecticut Yankee. This initial approach is being modified by ENRESA as described later in this section.

ENRESA utilized the latest RESRAD code (Version 6.4 Probabilistic) to determine Soil DCGLs for José Cabrera. In agreement with NRC guidance, to insure that the code yields conservative results, the user chooses conservative parameters based on the results of a sensitivity analysis that is performed by the RESRAD code. As the future use of the José Cabrera site is industrial, the dose model was based on the Industrial Worker Scenario.

Land Areas - Soil DCGLs

ENRESA evaluated a three cases in developing the José Cabrera Soil DCGLs:

- Industrial Worker Scenario to be used for the License Termination
- A Resident Farmer Scenario at specified times following license termination (as was done at Rancho Seco)
- Loss of Institutional Control as required by Spanish regulation – In this analysis the Resident Farmer Scenario is assumed at the time of License Termination and it is also assumed that the site has been remediated to the "Industrial Worker DCGL" developed per the first bullet.

The following are some of theimportant details of these three cases as applied to the José Cabrera site

Industrial Worker Scenario

For the Industrial Worker Scenario the following dose pathways were eliminated from the Soil DCGLs calculation:

- Plant Ingestion

- Meat Ingestion
- Drinking Water
- Ingestion of Aquatic Foods (Fish, etc.)

The following key input parameters were used in the RESRAD runs:

- No groundwater contamination at time = 0
- Occupancy factor: 2,000 hours per year
  - 74% of his/her time indoors while onsite
  - 26% of his/her time outdoors while onsite

The following site specific parameter data was used:

- A contaminated sand layer 15 centimeters (0.5 feet) thick
- Radionuclide concentrations used for the calculation from a single soil sample that represented the most highly contaminated soil measured at José Cabrera as indicated by gamma isotopic analysis.
- Seven of the potentially present radionuclides were detected in this sample. The sample results for the detected radionuclides were decayed from the date of analysis to the approximate completion date of Final Status Survey (FSS), May 2015.
- The 24 radionuclides that were not detected were evaluated based on the minimum detectable concentration (MDC) value for the sample. A decay period was applied to these MDC values to correspond to the approximate FSS completion date.

The following parameters were determined to be sensitive by RESRAD:

- Shielding Factor, external gamma
- Contaminant Zone Distribution Coefficient (K_d) for Cs-137

After running RESRAD, the Soil DCGL calculation was performed for the individual radionuclides using the following equation:

$$DCGL_i = 0.1 * CONC_i / DOSEPEAK_i \qquad \text{Eq. 4-4}$$

Where:

$DCGL_i$ = DCGL for soil for radionuclide i (Bq/g)
0.1 = Spanish Radiation dose limit of 0.1 mSv/yr (10 mrem/yr)
$CONC_i$ = Radionuclide concentration in soil sample decayed to May 2015 (Bq/g)
$DOSEPEAK_i$ = "Peak of the Mean" Dose determined by RESRAD for radionuclide i at concentration $CONC_i$ (mSv/yr)

Table 4.5 shows the José Cabrera Industrial Worker Soil DCGLs compared to the International Atomic Energy Agency (IAEA) Clearance Levels published in

Safety Guide, RS-G-1.7 (Reference 4-14). Table 4.5 shows that the industrial worker DCGLs are significantly higher than the IAEA Clearance Values.

Table 4-5
José Cabrera Industrial Worker Soil Derived Concentration Guidance Levels

Radio-nuclide	IAEA Clearance Level - Bq/g (pCi/g)	José Cabrera Industrial Worker Soil DCGLs at 0.1 mSv/yr - Bq/g (pCi/g)	Ratio - Industrial Worker DCGLs to IAEA Clearance Levels
Co-60	0.1 (2.7)	0.234 (6.32)	2.3 Times
Ni-63	100 (2700)	2.37 E+05 (6.41 E+06)	2.370 Times
Cs-134	0.1 (2.7)	0.417 (11.3)	4.2 Times
Cs-137	0.1 (2.7)	0.987 (26.7)	9.9 Times
Pu-239	0.1 (2.7)	105 (2,840)	1050 Times
Pu-240	0.1 (2.7)	105 (2,840)	1050 Times
Am-241	0.1 (2.7)	45.6 (1,230)	456 Times

Alternative Scenarios for Soil DCGLs

As mentioned above two additional scenarios were analyzed by ENRESA in evaluating José Cabrera's potential site release limits:

- Alternate Scenario 1: The Resident Farmer Scenario was used to evaluate the dose in future years from site land areas remediated to the limits determined using the Industrial Worker Scenario (Dose Limit: 0.1 mSv/yr [10 mrem/yr]). This calculation was made to understand and justify the timeframe over which the utility would keep control over the site to maintain industrial use.
- Alternate Scenario 2: As required by the Spanish regulations, this scenario assumes that the institutional controls fail. In this case the dose to the member of the critical group cannot exceed 1 mSv/year (100 mrem/yr) using an unrestricted scenario. This calculation was made to understand whether DCGLs for the Industrial Use Scenario or DCGLs assuming loss of institutional controls would be more conservative. The more conservative value would be chosen as the proposed soil DCGLs for the site.

For these scenarios, the normal Resident Farmer Scenario dose pathways are included as follows:

- Direct exposure to external radiation from the contaminated soil
- Internal dose from inhalation of airborne radionuclides
- Internal dose from ingestion of:
  - Contaminated soil
  - Plant foods grown in the contaminated soil and irrigated with contaminated water

- Drinking water from a contaminated well, and
- Meat and milk from livestock fed with contaminated fodder and water

Site Specific Input Parameters from the José Cabrera Offsite Dose Calculation Manual were used for the yearly intake quantities.

Sensitive Input Parameters determined for Resident Farmer Scenario

In performing RESRAD runs for the Resident Farmer Scenario at José Cabrera, the following were determined to be the sensitive input parameters:

- External gamma shielding factor
- Plant transfer factor for Cesium
- Depth of roots
- Meat transfer factor for Cesium
- Depth of soil mixing layer
- Contaminated zone distribution coefficient for Cs-137
- Milk transfer factor for Cesium.

Dose Impact from Alternative Scenarios

To evaluate the dose impact from Alternate Scenarios the following was done.

For Alternate Scenario 1, first the ratios of the radionuclides detected in the worst case soil sample and the single nuclide Industrial Worker DCGLs where used to determine the maximum allowable concentration for each radionuclide that would result in an annual dose to the industrial worker under the industrial worker scenario of 0.1 mSv/yr (10 mrem/yr). Next these maximum allowable concentrations under the Industrial Worker Scenario were used in RESRAD to calculate the annual dose to a future user assuming the Resident Farmer Scenario at various times after license termination. The results of this calculation are shown in Figure 4-2.

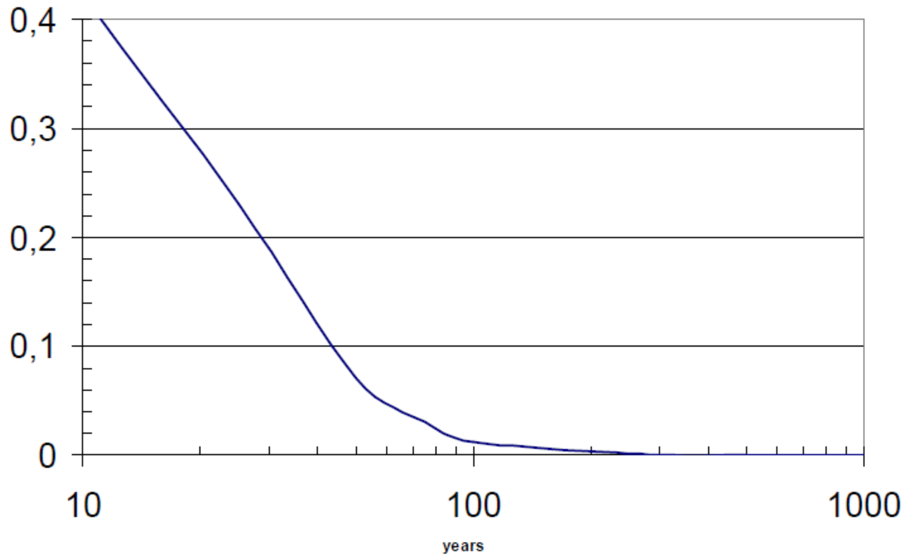

Figure 4-2
Resident Farmer Scenario Dose (in mSv/yr) when using José Cabrera Industrial Worker Soil DCGL Concentrations

This calculation (Figure 4-2) illustrates that if the site were remediated to the Industrial Worker DCGLs, the dose to the future user under the Resident Farmer Scenario, would equal the unrestricted release limit of 0.1 mSv/yr (10 mrem/yr) at approximately 43 years after license termination.

For Alternate Scenario 2, first RESRAD was run so as to calculate the single nuclide Resident Farmer DCGLs except that the dose criteria is changed to 1 mSv/yr (100 mrem/yr) to reflect the Spanish regulatory limit assuming the loss of institutional controls. The Soil DCGLs determined for the Alternate Scenario 2 were next compared to the Industrial Worker DCGLs that were originally calculated. The most limiting value is then used for each radionuclide as the soil DCGLs for the site.

Table 4-6 shows the comparison of the DGCL results and the limiting values for each of the key radionuclides.

Table 4-6
Comparison of José Cabrera Alternate Scenario to Industrial Worker Scenario and IAEA Clearance Values

Key Radio-nuclide	Alternate Scenario 2 Resident Farmer DGCL for 1 mSv/yr in Bq/g (pCi/g)	Industrial Worker DGCL for 0.1 mSv/yr in Bq/g (pCi/g)	Alternate Scenario 2 Resident Farmer to Industrial Worker DGCL (%)	Limiting DCGL in Bq/g (pCi/g)	IAEA Clearance Levels in Bq/g (pCi/g)	Ratio of Limiting DCGL to IAEA Clearance Values
Am-241	60.5 (1,640)	***45.6* (1,230)***	133	45.6 (1,230)	0.1 (2.7)	456 Times
Co-60	0.654 (17.7)	***0.234* (6.32)***	280	0.234 (6.32)	0.1 (2.7)	2.34 Times
Cs-134	1.03 (27.8)	***0.417* (11.3)***	247	0.417 (11.3)	0.1 (2.7)	4.17 Times
Cs-137	2.22 (60)	***0.987* (26.7)***	225	0.987 (26.7)	0.1 (2.7)	9.87 Times
Ni-63	***95,900* (2.6 E+06)***	23,700 (6.41 E+06)	40.5	95,900 (2.6 E+06)	100 (2,700)	959 Times
Pu-239	***67.5* (1,820)***	105 (2,840)	64.3	67.5 (1,820)	0.1 (2.7)	675 Times
Pu-240	***67.7* (1.830)***	105 (2,840)	64.4	67.7 (1.830)	0.1 (2.7)	677 Times

*Limiting Values

As can be seen from Table 4-6, the limiting DCGLs for Ni-63, Pu-239 and Pu-240 are the Alternative Scenario 2 values. These values are based on the loss of institutional control resulting in the Resident Farmer Scenario becoming applicable and a dose limit of 1 mSv/yr (100 mrem/yr). However, the DCGL concentrations for these three radionuclides are not expected to cause additional soil remediation. The DCGLs values for these radionuclides are much higher than the concentrations for these radionuclides that have been present at power plant sites that have been remediated to the U.S. site release criteria of 25 mrem/yr (0.25 mSv/yr). Experience shows that, typically, the concentrations of Co-60 and/or Cs-137 drive the remediation at power plant sites.

Land Areas - Groundwater DCGLs

As mentioned above, ENRESA calculated Groundwater DCGLs for José Cabrera in a manner similar to what was done at Connecticut Yankee (see Section 4.2.2.1.1). The Resident Farmer Scenario was used with the following dose pathways included:

- Plant foods irrigated with water containing residual radioactivity
- Meat and milk from livestock fed with water containing residual radioactivity; and
- Drinking water containing residual radioactivity from a well

The José Cabrera model assumes that the groundwater contains residual radioactivity at the time of site release and that all sources that contributed to this contamination have since been removed. Specific Site Parameter Data from the José Cabrera Offsite Dose Calculation Manual was also used.

Other key input parameters were set as follows:

- Time since placement of material = 1 year
- Time for calculation = 1 year
- Model for water transport parameters= Mass Balance (MB) model
- Distribution coefficient in the saturated zone= 0 cm3/g

The equilibrium groundwater concentration associated with the contaminated zone was calculated as described in the "Users Manual for RESRAD Version 6.0" (See Section 4.2.2.1.1).

Consistent with the guidance of Argonne National Laboratories (RESRAD developer), certain input parameters need to be set as follows when modeling groundwater contamination in the RESRAD runs:

- An initial radionuclide concentration of 1 Bq/g (27 pCi/g) was used for the soil in the contaminated zone.
- For Pu-239/240, Pu-241, Am-241 and Cm-243/244, the groundwater DCGL were calculated by modeling the decay of a unit source over 1000 yrs.

- Table 4-7 shows the Groundwater DCGLs determined for José Cabrera by the process described above.

Table 4-7
José Cabrera Groundwater Derived Concentration Guidance Levels

Radionuclide	Groundwater DCGLs using Resident Farmer and 0.1 mSv/yr limit in Bq/l (pCi/l)	José Cabrera Groundwater Concentrations as of May 2009 in Bq/l (pCi/g)	Groundwater Dose at the Groundwater Concentrations as of May 2009 in mSv/yr (mrem/yr)
H-3	6,300 (170,000)	4,300 (116,000)	0.0687 (6.87)
C-14	1,830 (49,500)	9.6 (259)	0.00053 (0.053)
Co-60	19.3 (521)	Non-detectable	0
Sr-90	3.24 (87.6)	0.17 (4.59)	0.00525 (0.525)
Cs-134	2.87 (77.6)	Non-detectable	0
Cs-137	3.6 (97.3)	Non-detectable	0

Table 4-7 shows that the dose from the present contamination in groundwater for most nuclides (not including tritium, H-3) is less than or equal to 6% of the total allowable dose for soil and groundwater at José Cabrera. Although the dose from H-3 corresponding to the May 2009 concentrations is a large percentage of the site release criteria, it is important to note that these are pre-remediation concentrations. Once remediation has removed the source of the tritium, it is expected that natural attenuation will reduce the H-3 groundwater concentrations relatively quickly (i.e., during the industrial use period of the site) since the radionuclide is very mobile (moves at the same rate as groundwater).

Future Work for José Cabrera

The work that has been described to date was to develop the preliminary DCGLs for the José Cabrera site. Additional evaluations of the DCGLs at José Cabrera are now in progress as follows:

- Sensitivity analysis are being performed for each individual radionuclide potentially present at the site in measurable quantities, not just the mixture of radionuclides based on the worst case site soil sample
- Statistical parameter distribution will be input into RESRAD for performing sensitivity analyses, only for physical parameters ranked as priority 1 or 2. The following parameters may be used with a deterministic (fixed) value:
  - Inhalation rate
  - Soil ingestion rate
  - Depth of roots

- Instead of assigning 25% or 75% Quartile values to Positive/Negative correlation as was done for Rancho Seco (as described in Section 4.2.2.1.2) parameter distributions will be assigned to all RESRAD parameters covered by the last bullet when performing the RESRAD DCGL calculations
- In agreement with the Regulatory Body (CSN) comments to the previous approach, a revised approach, using the modifications described in the bullets above will be undertaken. The resulting calculations will be submitted to the Regulatory Body upon completion.

Summary of José Cabrera DCGL Development

ENRESA has utilized the applicable experiences at other sites as a basis for the development of preliminary DCGLs for the José Cabrera site. It is expected that the DCGLs developed through this dose based and site specific analysis will reduce the remediation requirements that may have been required using other release limits (i.e., IAEA Clearance Levels) while adequately protecting the future user of the site.

4.2.2.1.4 Yankee Rowe Experience

Yankee Rowe used the RESRAD code to calculate the soil DCGLs for the Resident Farmer Scenario. For all RESRAD code runs for Yankee Rowe, the "Peak of the Mean" (highest dose value over all of the years for which the analysis is performed) was used to determine the 25 mrem/yr (0.25 mSv/yr) soil DCGLs. More detailed information on dose modeling with RESRAD is provided in a separate EPRI Report (Reference 4-1). Table 4-8 shows the 25 mrem/yr (0.25 mSv/yr) soil DCGLs for Yankee Rowe.

Table 4-8
Yankee Rowe Soil Derived Concentration Guidance Limits

Radionuclide	Soil DCGL at 25 mrem/year (0.25 mSv/yr) in pCi/g (Bq/g)	Soil DCGL 8.73 mrem/year (0.0873 mSv/yr) in pCi/g (Bq/g)
H-3	372 (13.8)	130 (4.8)
C-14	5.4 (0.2)	1.9 (0.07)
Fe-55	28,600 (1,060)	9,990 (370)
Co-60	4.0 (0.15)	1.4 (0.05)
Ni-63	802 (30)	280 (10.5)
Sr-90	1.7 (0.064)	0.59 (0.022)
Nb-94	7.2 (0.26)	2.5 (0.091)
Tc-99	14 (0.51)	4.9 (0.18)
Ag-108m	7.2 (0.26)	2.5 (0.091)
Sb-125	32 (1.17)	11 (0.41)
Cs-134	4.9 (0.18)	1.7 (0.063)
Cs-137	8.6 (0.32)	3.0 (0.11)
Eu-152	10 (0.37)	3.5 (0.13)
Eu-154	9.5 (0.35)	3.3 (0.12)
Eu-155	401 (15)	140 (5.2)
Pu-238	32 (1.17)	11 (0.41)
Pu-239/240	29 (1.06)	10 (0.37)
Pu-241	974 (36)	340 (0.12.6)
Am-241	29 (1.06)	10 (0.37)
Cm-243/244	32 (1.17)	11 (0.41)

The site had committed to achieve the Groundwater Maximum Contaminant Levels (MCLs) for radionuclides in groundwater. The dose that corresponded to these MCL concentrations was 0.77 mrem/yr (0.0077 mSv/yr). As such, 0.77 mrem/yr (0.007 mSv/yr) of the 25 mrem/yr (0.25 mSv/yr) limit needed to be allocated to doses from "existing" groundwater contamination. This value corresponds to the dose from 20,000 pCi/L (740 Bq/l) of H-3 in groundwater calculated using RESRAD with the Resident Farmer scenario and utilizing many parameters specific to the Yankee Rowe site. Yankee Rowe also calculated the dose from the activity projected to leach out of subsurface structures into groundwater to be 0.5 mrem/yr (0.005 mSv/yr).

Additionally, Yankee Rowe needed to comply to the State of Massachusetts Department of Public Health regulations: 10 mrem/year (0.1 mSv/yr) TEDE plus ALARA. The 25 mrem/year (0.25 mSv/yr) Soil DCGLs were adjusted down to 8.73 mrem/year (0.0873 mSv/yr). This resulted in a total dose from all media of 10 mrem/year (0.1 mSv/yr) when the dose allotted to subsurface structures and groundwater was included. The adjusted soil DCGL values are shown in Table 4-8.

4.2.2.1.5 Maine Yankee Experience

Determining Derived Concentration Guidance Levels for Soil

As previously discussed, Maine Yankee (MY) committed to a release criteria of 10 mrem/yr (0.1 mSv/yr) TEDE from all pathways with no more than 4 mrem/yr (0.04 mSv/yr) TEDE from the groundwater drinking source (Reference 4-15). The 10 mrem/yr (0.1 mSv/yr) site release criteria were set in agreement with the State of Maine. In a similar manner to Connecticut Yankee (CY), doses due to various media and pathways were determined so that the sum of the doses did not exceed the 10 mrem/yr release criteria Unlike CY, the target dose for the various pathways was pre-set at the time of LTP submittal based on characterization data and dose analyses. It was possible to set these target levels early as Maine Yankee characterization data generally showed little groundwater contamination and did not show significant levels of radionuclides other than tritium in groundwater. Maine Yankee needed to adjust these pre-set values a few times during the decommissioning as additional dose sources were identified or dose modeling approaches revised. In all cases, Maine Yankee was able to reach agreement to these changes with the various stakeholders. Tables 4-9 and 4-10 list the final DCGLs for the various contaminated materials and the annual dose that corresponds to these area and media specific DCGLs.

The general approach was to set the Soil DCGLs after doses from all the other media had been subtracted from the 10 mrem/yr (0.1 mSv/yr) release criteria. The dose from media other than soil at Maine Yankee was determined to be no more than 0.72 mrem/yr (0.0072 mSv/yr), leaving 9.28 mrem/yr (0.0928 mSv/yr) for the soil. Once the dose allotted for soil was determined the corresponding surface and deep (subsurface) soil DCGLs were calculated using the ratio of the NRC Soil Screening Values (corresponding to 25 mrem/yr (0.25 mSv/yr) as discussed in Section 4.2.1). For deep soil, the direct dose was calculated using the Microshield Code, assuming 6 inches (15 cm) of soil for shielding. Another dose pathway from deep soil is due to the drinking water pathway. The RESRAD Code (discussed above for CY) was used to calculate the dose due to contaminated groundwater in contact with the deep soil.

Table 4-9
Maine Yankee Contaminated Material Derived Concentration Guidance Levels

Material	DCGL
Basement Contaminated Concrete (gross beta dpm/100 cm^2)	18,000 (300 Bq/cm^2)
Special Area Contaminated Concrete (gross beta dpm/100 cm^2)	9,500 (160 Bq/cm^2)
Basement Activated Concrete Total Inventory (pCi)	4.88 E+08 (18 MBq)
Surface Soil and Surface Soil (pCi/g) (Cs-137)*	2.39 (0.088 Bq/g)
Balance of Plant (BOP) Embedded Piping (gross beta dpm/100 cm^2)	100,000 (1,700 Bq/cm^2)
Spray Building Pump Embedded Piping (gross beta dpm/100 cm^2)	800,000 (13,000 Bq/m^2)
Ground Water (H-3, pCi/L)	6,812 (252 Bq/l)
Surface Water (H-3, pCi/L)	960 (35.5 Bq/l)
Buried Piping, Conduit and Cable (gross beta dpm/100 cm^2)	9,500 (352 Bq/g)

*Note: Soil DCGLs based on 6.77 mrem/year (0.067 mSv/yr) total dose from surface & sub-surface soil.

Table 4-10
Maine Yankee Contaminated Material Annual Dose

Material	Drinking Water mrem/yr (mSv/yr)	Direct, Inhalation & Ingestion mrem/yr (mSv/yr)	Total Annual Dose mrem/yr (mSv/yr)
Basement Concrete	2.7 E-01 (2.7 E-03)	3.08 E-02 (3.08 E-04)	3.01 E-01 (3.01 E-03)
Basement Activated Concrete	1.05 E-02 (1.05 E-04)	3.02 E-02 (3.02 E-04)	4.06 E-02 (4.06 E-04)
Surface Soil	0.00 E+00	5.63 (5.63 E-02)	5.63 (5.63 E-02)
Subsurface Soil	2.97 E-02 (2.97 E-04)	1.11 (1.11 E-02)	1.14 (1.14 E-02)
BOP Embedded Piping	4.59 E-02 (4.59 E-04)	5.23 E-03 (5.23 E-05)	5.11 E-02 (5.11 E-04)
Spray Building Embedded Piping	7.6 E-02 (7.6 E-04)	8.67 E-03 (8.67 E-05)	8.47 E-02 (8.47 E-04)
Ground Water	2.08 E-01 (2.08 E-03)	0.00 E+00	2.08 E-01 (2.08 E-03)
Surface Water	2.94 E-02 (2.94 E-04)	1.27 E-03 (1.27 E-05)	3.06 E-02 (3.06 E-04)
Buried Piping, Conduit and Cable	6.33 E-04 (6.33 E-06)	1.89 E-03 (1.89 E-05)	2.52 E-03 (2.52 E-05)
Totals	6.8 E-01 (6.8 E-03)	6.81 (6.81 E-02)	7.49 (7.49 E-02)

The soil DCGLs were further reduced so that the total dose from all media would be less than 10 mrem/yr (0.1 mSv/yr). The final MY License Termination Plan implemented a DCGLs that corresponded to 6.77 mrem/yr (0.067 mSv/yr) from soil and 0.72 mrem/yr (0.0072 mSv/yr) from all other pathways, resulting in a total dose of 7.49 mrem/yr (0.075 mSv/yr). This total dose is well below the 10 mrem/yr limit to provide a margin if media characterization found higher concentrations of radionuclides in media during the final status survey conducted after the release limits had been approved.

Dose from Ground Water and other Water Related Dose Pathways

Maine Yankee performed a drinking water dose calculation using the highest concentration of tritium (H-3) that had been measured in groundwater. Maine Yankee also performed dose calculations for drinking water and ingestion of fish from an on-site pond using the highest concentration of tritium (H-3) that had been measured in surface water. This concentration and the calculated dose are shown in Tables 4-9 and 4-10. These doses were included in the sum of the doses from all other media that was used in the calculation of soil DCGLs.

4.2.2.1.6 Big Rock Point

The Big Rock Point Plant utilized a combination of DCGLs to perform final status surveys. For areas with medium or low potential for concentrations approaching DCGLs, the NRC soil screening values were utilized. For areas with high potential for exceeding DCGLs prior to remediation, site specific DCGLs (approved in the Big Rock Point LTP by the NRC) were utilized. Table 4-11 shows the NRC screening values and the Big Rock Point site specific soil DCGLs for selected key radionuclides. A review of the table shows that the site specific DCGLs, developed using the modified Resident Farmer Scenario (discussed in Section 3.3.5) are higher than the screening values for most radionuclides. However, the site specific DCGL for Co-60 is actually lower than the screening value due to the different computer codes used to determine each value.

Table 4-11
Big Rock Point Soil Release Limits

Key Radionuclide	Site Specific DCGL [Used for Class 1 Areas] pCi/g (Bq/g)	NRC Screening Values [Used for Class 2 and 3 Areas] pCi/g (Bq/g)
H-3	327 (12.1)	110 (4.1)
Fe-55	358,000 (13,200)	10,000 (370)
Co-60	3.21 (0.12)	3.8 (0.14)
Cs-137	13.2 (0.49)	11 (0.41)

4.2.2.1.7 Saxton

The Saxton Nuclear Experimental Corporation Plant was a 23.5 MWe PWR designed by Westinghouse, Corporation and began commercial operation in 1962. It was permanently shutdown in 1972 and remained in a SAFSTOR condition until 1998 when decommissioning began.

At Saxton site-specific DCGLs for soil (both surface and subsurface) were developed assuming the Resident Farmer Scenario and using the RESRAD code. Site-specific K_ds were determined and used in the RESRAD code runs. For comparison to the DCGLs at other plants, the Saxton 25 mrem/yr (0.25 mSv/yr) soil DCGL for Cs-137 was 8.5 pCi/g (0.32 Bq/g.)

4.2.2.1.8 San Onofre Unit 1(SONGS-1)

The SONGS-1 plant has proceeded with much of its decommissioning without having submitted an LTP to the NRC. Soil contamination at SONGS-1 has been remediated to the NRC soil screening values. It is the SONGS-1 policy that by remediating the site soils to the conservative screening values any hypothetical contamination of groundwater would be below the EPA MCLs.

4.2.2.2 Dose Modeling for Sediments

If radionuclide contamination is found in underwater sediments at a site, the licensee may need to determine whether remediation of the material may be required. The following provides a summary of experiences addressing contaminated sediments during power plant site decommissioning.

4.2.2.2.1 Connecticut Yankee Discharge Canal and Permanent Wetlands

The final status surveys required for the Connecticut Yankee (CY) site included the area of the one mile (1.7 km) long discharge canal. Compliance with the License Termination Plan (LTP) required that the sediments be sampled and meet the Soil DCGLs discussed in Section 4.3.2.1.1. This is considered appropriate given that the action that would result in the greatest dose to future inhabitants of the site would be to dredge up the sediment, place it on land and use it for farming. Dose to a future site occupant would be much lower if they did not use the sediment in this way, since many of the significant pathways considered in developing the soil DCGLs (i.e. direct exposure, uptake by plants, etc.) would not apply.

4.2.2.2.2 Maine Yankee Forebay and Diffuser

The Maine Yankee (MY) Forebay was a fairly large man made, rock lined, rectangular pond into which the circulating water and plant liquid radioactive effluents were discharged (Figure 4-3). After transiting through the forebay, discharged water would pass through a diffuser pipe and into the Back River (which was influenced by the tide). The bottom consisted of rock, gravel, and ledges covered by marine sediment. The sediment layer ranged from inches to feet (several centimeters to several hundred centimeters) in thickness. MY performed an evaluation of potential for dose from the contamination contained in the forebay sediments. This dose evaluation was performed even though no contamination had been measured in the surface water in the forebay and the brackish nature of the water makes consumption and irrigation doubtful. The dose from surface water was determined to be insignificant and therefore not added to the doses in Table 4-10.

Figure 4-3
Maine Yankee Forebay Before and During Remediation and After Final Status Survey

The dose scenario modeled the forebay as a backfilled basement and resulted in the following DCGLs:

- For the rock to remain – 18,000 dpm/100cm^2 (300 Bq/m^2)
- For sediment to remain – 7 pCi/g [2.59 E (-01) Bq/g] (Small areas could contain higher levels if passing the Elevated Measurement Comparison Test defined in MARSSIM, Reference 2-2)

The Maine Yankee (MY) Plant Forebay needed to be remediated to meet these DCGLs. The remediation required 4 months to complete and involved underwater vacuum removal of essentially all of the marine sediment (Figure 4-3). This was followed by the FSS which required two months to complete. Details on this survey are contained in the EPRI In-situ Gamma Spectroscopy Report (Reference 1-5).

4.2.2.3 Summary of Plant Specific Dose Modeling Experience for Land Areas

This section provides a summary of the plant specific dose modeling experience for land areas along with a comparison to NRC Soil Screening Values. Table 4-12 provides a comparison of Soil Derived Concentration Guidance Levels (DCGLs) determined by different computer codes and using different dose modeling scenarios. By reviewing the ratio of the Connecticut Yankee (CY) site specific values, as an example, to the NRC Screening Values in Table 4-12 the following can be concluded:

- Most of the gamma emitting radionuclides (except Cs-134 and Cs-137 due to their greater mobility through the soil to the groundwater) have the same, or somewhat higher, values for the CY site specific vs. the NRC screening values.
- Most of the Beta-only emitting radionuclides have lower DCGLs for CY versus the NRC screening values due to their greater mobility through the soil to the groundwater (and resulting dispersion of concentrations). However, as H-3 is treated as tritiated water in both the CY and the NRC screening values calculations, the more conservative inputs to the D&D code make the NRC screening values lower for H-3.
- All of the Transuranic radionuclides such as Am-241 and Pu-238 have much higher site specific DCGLs than the NRC screening values due to the radionuclide specific input parameters used in the CY analysis.

The last column of Table 4-12 shows soil DCGLs calculated by RESRAD using the key default parameters of a contaminated zone thickness of 2 meters and a depth to groundwater of 4 meters. It is noteworthy that although the Cs-137 "RESRAD Default DCGL" is nearly the same as the NRC screening values, the "RESRAD Default DCGLs" for Sr-90 and Pu-239 are much higher than the corresponding NRC screening values. This is due to the RESRAD default parameters being less conservative in some cases then the D&D code input parameters.

The conclusion that can be made from chapter is that the calculation of Resident Farmer DCGLs via a site specific analysis can result in higher concentrations than the NRC soil screening values but that the results are highly dependent on the characteristics and the future use of the site be evaluated.

Table 4-12 also contains the corresponding Site Specific Soil DCGLs determined for Rancho Seco using the Industrial Worker Scenario. The table shows considerably higher values for Rancho Seco than those determined using the Resident Farmer Scenario for the NRC Screening Values or the CY Site-specific Soil DCGLs. These results illustrate that defensible controls on the future use of the site can allow for higher release limits.

Table 4-12
Comparison of Site Specific Soil Derived Concentration Guidance Levels (DCGLs) to Generic Soil DCGLs

Radio-nuclide	NRC Screening Values (Resident Farmer Scenario) pCi/g (Bq/g)	CY Site Specific DCGLs (RESRAD Code using Resident Farmer Scenario) pCi/g (Bq/g)	Rancho Seco Industrial Worker DCGLs (RERAD) pCi/g (Bq/g)	Resident Farmer DCGLs: RESRAD Using Default Input Parameters (Reference 4-2) pCi/g (Bq/g)
H-3	1.1 E+02 (4.1)	4.12 E+02 (15.2)	Not Required	N/A
C-14	1.2 E+01 (0.44)	5.66 E+00 (0.21)	8.33E+06 (3.08E+04)	N/A
Mn-54	1.5 E+01 (0.55)	1.74 E+01 (0.64)	Not Required	N/A
Fe-55	1.0 E+04 (369)	2.74 E+04 (1,014)	Not Required	N/A
Co-60	3.8 E+00 (0.14)	3.81 E+00 (0.14)	1.26E+01 (0.47)	N/A
Ni-63	2.1 E+03 (77)	7.23 E+02 (26.8)	1.52E+07 (5.62E+05)	N/A
Sr-90	1.7 E+00 (0.06)	1.55 E+00 (0.06)	6.49E+03 (240)	4.91 (0.18)
Nb-94	5.8 E+00 (0.21)	7.12 E+00 (0.26)	Not Required	N/A
Tc-99	1.9 E+01 (0.70)	1.26 E+01 (0.47)	Not Required	N/A
Cs-134	5.7 E+00 (0.21)	4.67 E+00 (0.17)	2.24E+01 (0.83)	N/A
Cs-137	1.1 E+01 (0.41)	7.91 E+00 (0.29)	5.28E+01 (1.95)	10.9 (0.40)
Eu-152	8.7 E+00 (0.32)	1.01 E+01 (0.37)	Not Required	N/A
Eu-154	8.8 E+00 (0.33)	9.29 E+00 (0.34)	Not Required	N/A
Pu-238	2.5 E+00 (0.09)	2.96 E+01 (1.10)	Not Required	N/A
Pu-239	2.3 E+00 (0.08)	2.67 E+01 (0.99)	Not Required	56.9 (2.11)
Pu-241	7.2 E+01 (2.6)	8.70 E+02 (32.2)	Not Required	N/A
Am-241	2.1 E+00 (0.08)	2.58 E+01 (0.95)	Not Required	N/A
Cm-243	3.2 E+00 (0.12)	2.90 E+01 (1.07)	Not Required	N/A

"Not Required": For Rancho Seco, these were shown not to have the potential for significant contribution to the post closure dose at the site. Per NRC guidance (See Section 4.2.2.1.2, DCGL development was not required as analysis for these radionuclides was not required.
"N/A": Not Available

4.3 Adjusting Site Release Limits For Multiple Contaminated Mediums

As the site release criteria applies to all applicable dose pathways in the area being evaluated, the presence of multiple contaminated medium needs to be considered. An example of this is where sample results for a site show contamination in groundwater (called "existing" groundwater) and contamination in the soil above the groundwater. In this case the current groundwater concentrations are normally compared to the groundwater DCGLs and a yearly dose due to "existing" groundwater is calculated. The "existing" groundwater dose is then subtracted from the site release criteria (i.e., 25 mrem/year [0.25 mSv/yr]) to determine the dose that remains that can be applied to the soil in the unsaturated zone of the contaminated area.

Section 4.2.2.1.1 illustrates how this allotment of dose was performed during the Connecticut Yankee Plant decommissioning.

Section 5: Development of Site Release Limits for Structures

5.1 Overview

The evolution of site release limits for buildings was similar to that described in Chapter 4 for land areas. The early U.S Nuclear Regulatory Commission (NRC) guidance primarily described one scenario: the Building Occupancy Scenario. Updated NRC guidance published in the 2003-2004 timeframe described the use of "realistic" scenarios. This chapter will first describe the Building Occupancy Scenario and then cover site-specific experiences with other scenarios.

This chapter contains a summary of the dose modeling experiences for concrete structures. More details on the dose calculation methods used at the different decommissioning sites is contained in a separate EPRI Report, “Concrete Characterization and Dose Modeling During Plant Decommissioning: Detailed Experiences 1993 – 2007” (EPRI Report 1015502, Reference 1-3).

5.2 Building Occupancy Scenario

In this scenario, occupants of a structure are assumed to be office workers that spend 40 or more work hours each week in a room that was formally part of the plant. This room is assumed to have been released for use as an office. The D&D and RESRAD-Build computer codes are commonly used to calculate these building DGCLs.

5.3 Options for the Development of Site Release Limits

5.3.1 NRC Published Screening Values for Structures

The NRC has published screening values for residual radioactivity in structures. These screening values are provided in “Federal Register, 63 FR 64132, Supplemental Information on the Implementation of the Final Rule on Radiological Criteria for License Termination” (Reference 5-1). Table 5-1 provides screening values for most of the radionuclides normally detected during power plant decommissioning. The screening values were calculated by the D & D code using the conservative default input parameters. The use of the default parameters in the D & D code resulted in conservative release limits, these limits were considered “pre-approved” by the NRC [i.e. licensee did not need to gain approval as part of an License Termination Plan (LTP) submittal]. Limitations on the use of the generic screening values for building surfaces as given in NUREG-1757, "Consolidated Decommissioning Guidance", Volume 2, Rev. 1 (Reference 2-1) are as follows:

- Contamination on building surfaces should be surficial and not-volumetric [no more than 10 mm (0.39 in) of penetration]
- Residual Radioactivity on the building surface is mostly fixed (loose residual radioactivity no more than 10% of total surface activity)
- Use of the screening values on buried structures (e.g., drainage or sewer pipes) or equipment within the building requires justification and evaluation by the NRC on a case-by-case basis.
- It should be noted that the screening values or Derived Concentration Guidance Levels (DCGLs) for structures are for individual radionuclides. When mixtures of radionuclides are present the "Sum of Fractions" rule must be applied. NRC also noted in NUREG-1757 (Reference 2-1) that the use of the single default parameter set for all radionuclides in developing the screening DCGLs (as was done in calculating the NRC screening values) could result in overly conservative limits. The user is instructed that tailoring the default parameter set to individual radionuclides would, in most cases, result in higher DCGLs. Industry experience has shown this effect to be particularly evident for the alpha-emitting radionuclides.

5.3.2 Site-Specific Dose Modeling Experience

As with the development of site release criteria for land areas discussed above, the situation at Connecticut Yankee (CY) necessitated a more complex method for calculating building DCGLs to show compliance with the unrestricted release limits than many other facilities. The CY experience will be discussed first followed by experiences at other plants. Most of the information in this section concerning CY was obtained from References 4-6 and 5-2.

5.3.2.1 Connecticut Yankee Experience

Connecticut Yankee performed calculations using three different dose models to determine site release limits for concrete over the course of the decommissioning. The approaches used by CY were the Building Occupancy Scenario, Concrete Debris Scenario, and Basement Fill Model. Each is described in more detail in the sections below.

5.3.2.1.1 Building Occupancy Scenario

The conceptual model underlying Building Occupancy Scenario dose model for CY consisted of a room of fixed area [10 meter (m) by 10 m by 2.5 m high], uniform concentrations of residual radioactivity on all room surfaces, and the receptor located at the center of the room at a height of 1 m. Two cases were considered for the source type: area (surface) sources and volume sources. Area sources consisted of a thin-layer of residual radioactivity on the surface, consistent with NUREG/CR-5512, Volume 1, "Residual Radioactive Contamination From Decommissioning: Technical Basis for Translating Contamination Levels to Annual Total Effective Dose Equivalent," (Reference 5-3). Volumetric sources consisted of concrete to the depth of 0.305 m (12 inches) to account for the possibility of contamination within the concrete. The volume of concrete could have been contaminated by migration of radioactive material into the depth of the source or by neutron activation.

Table 5-1
CY Building Surface Derived Concentration Guidance Levels Compared to NRC Values.

Radio-nuclide	CY Site Specific DCGL (dpm/100 cm^2) (Note 2)	NRC Screening Values (dpm/100 cm2) (Note 2)	Ratio: Site Specific DCGLs to Screening Values (Times Higher)
H-3	3.15 E+08	1.2 E+08	2.6
C-14	1.03 E+07	3.7 E+06	2.8
Mn-54	3.21 E+04	3.2 E+04	1
Fe-55	3.49 E+07	4.5 E+06	7.8
Co-60	1.11 E+04	7.1 E+03	1.6
Ni-63	3.60 E+07	1.8 E+06	20.0
Sr-90	1.27 E+05	8.7 E+03	14.6
Nb-94	1.71 E+04	Note 1	N/A
Tc-99	1.45 E+07	1.3 E+06	11.2
Ag-108m	1.65 E+04	Note 1	N/A
I-129	Not of Concern at CY	3.5 E+04	N/A
Cs-134	1.65 E+04	Note 1	N/A
Cs-137	4.30 E+04	2.8 E+04	1.5
Eu-152	2.34 E+04	Note 1	N/A
Eu-154	2.19 E+04	Note 1	N/A
Eu-155	4.37 E+05	Note 1	N/A
Pu-238	4.87 E+03	Note 1	N/A
Pu-239	4.44 E+03	Note 1	N/A
Pu-241	2.29 E+05	Note 1	N/A
Am-241	4.27 E+03	Note 1	N/A
Cm-243	6.07 E+03	Note 1	N/A

Notes:
1. Values are not published in the Federal Register but can be determined using the D & D code. 2. 1 dpm/100 cm^2 = 1.67 Bq/m^2

This scenario assumes that the building will be used as an office building after release of the site from the license. A dose model from the RESRAD family of codes, developed by Argonne Labs, called RESRAD-Build was used to determine the building DCGLs at CY. The same process of determining input parameters using the Probabilistic (Version 3.1) and Deterministic (Version 2.37) versions of the model for determining soil DCGLs (as was in Chapter 4) was used for the Building Occupancy DCGL determination.

Table 5-1 lists the site-specific 25 mrem/yr (0.25 mSv/yr) Building Occupancy DCGLs for Connecticut Yankee, as well as the NRC Generic Screening DCGLs published in the Federal Register. Most of the site specific values are significantly higher than the generic values. This illustrates the conservatism in the default input parameters used to derive the screening DCGLs.

5.3.2.1.2 Concrete Debris Scenario

In the early stages of the decommissioning, CY planned to demonstrate that the above ground concrete on site was acceptable for release even if they remained standing. The concrete structures would then be demolished into their basements and the debris would be covered with material that met the site release criteria. This scenario was called the Concrete Debris Scenario. For this case, the concrete debris was treated as soil when using the RESRAD code. The results are shown in Table 5-2 along with the volumetric DCGLs determined by using the Building Occupancy Scenario. To demonstrate compliance with post closure release limits the lower of the two DCCLs was to be used for concrete debris left on-site.

Table 5-2
CY DCGLs for Building Demolished (Concrete Debris Scenario)

Radionuclide	Concrete Debris DCGLs pCi/g (Bq/g)	DCGL for Volumetric Sources - Building Occupancy Scenario pCi/g (Bq/g)
H-3	9.05E+01 (3.3E+00 Bq/g)	1.47E+03 (5.4E+01 Bq/g)
C-14	2.05E+01 (7.6E-01 Bq/g)	1.18E+08 (4.4E+06 Bq/g)
Mn-54	5.51E+01 (2.0E+00 Bq/g)	9.06E+00 (3.3E-01 Bq/g)
Fe-55	8.96E+01 (3.3E+00 Bq/g)	9.54E+07 (3.5E+06 Bq/g)
Co-60	9.07E+01 (3.4E+00 Bq/g)	2.90E+00 (1.1E-01 Bq/g)
Ni-63	1.29E+02 (4.8E+00 Bq/g)	4.11E+07 (1.5E+06 Bq/g)
Sr-90	3.77E-01 (1.4E-02 Bq/g)	2.38E+03 (8.8E+01 Bq/g)
Nb-94	7.74E+00 (1.5E-01 Bq/g)	4.83E+00 (1.8E-01 Bq/g)
Tc-99	2.85E+01 (1.1E+00 Bq/g)	3.09E+07 (1.1E+06 Bq/g)
Ag-108m	2.59E+01 (9.6E-01 Bq/g)	4.84E+00 (1.8E-01 Bq/g)
Cs-134	3.21E+02 (1.2E+01 Bq/g)	4.93E+00 (1.8E-01 Bq/g)
Cs-137	6.45E+02 (2.4E+01 Bq/g)	1.37E+01 (5.1E-01Bq/g)
Eu-152	2.27E+02 (8.4E+00 Bq/g)	6.70E+00 (2.5E-01 Bq/g)

Table 5-2 (continued)
CY DCGLs for Building Demolished (Concrete Debris Scenario)

Radionuclide	Concrete Debris DCGLs pCi/g (Bq/g)	DCGL for Volumetric Sources - Building Occupancy Scenario pCi/g (Bq/g)
Eu-154	1.94E+02 (7.2E+00 Bq/g)	6.11E+00 (2.3E-01 Bq/g)
Eu-155	9.53E+03 (3.5E+02 Bq/g)	3.23E+02 (1.2E+01Bq/g)
Pu-238	1.14E+01 (4.2E+01 Bq/g)	6.61E+02 (2.4E+01 Bq/g)
Pu-239	1.00E+01 (3.7E-01 Bq/g)	6.02E+02 (2.2E+01Bq/g)
Pu-241	1.49E+02 (5.5E+00 Bq/g)	3.12E+04 (1.2E+03 Bq/g)
Am-241	4.42E+00 (1.6E-01 Bq/g)	4.16E+02 (1.5E+01 Bq/g)
Cm-243	3.83E+00 (1.4E-01 Bq/g)	7.53E+01 (2.8E+00 Bq/g)

5.3.2.1.3 Basement Fill Model

CY used the Basement Fill Model to determine the projected dose from the leaching of radionuclides from contaminated concrete. This calculated dose was the "Future Groundwater" dose component of the Equation 4-1, the CY compliance equation. As discussed above in Section 4.2.2.1.1, the "Future Groundwater" component might not apply in all situations. Connecticut Yankee submitted the Basement Fill Model for NRC approval as a "realistic" scenario. As discussed in Chapter 3, NRC encouraged the use of realistic scenarios, where appropriate, in its updated guidance.

CY made the decision, in 2004, to dispose of all of the above ground concrete as either clean (containing no detectable plant-related contamination) or radioactive waste. This decision made the Concrete Debris Scenario no longer applicable. In the same timeframe, the Maine Yankee Decommissioning Project had received approval for a scenario called the Basement Fill Model. In this model, basements for structures to remain on site were considered inaccessible. The radioactivity that remained in the concrete was assumed to leach from the concrete into groundwater and result in dose due to the groundwater pathway.

The scenario of the Basement Fill Model for CY is as follows:

- The radioactivity inventory in the concrete was assumed to diffuse out of the concrete surface. Conservative high values for diffusion rates were chosen from documented diffusion studies.
- The total amount of radioactivity released from all the below grade concrete in the containment and fuel pool was assumed to move to the inside of the containment basement.
- This radioactivity inventory in the containment basement was assumed to mix with the groundwater and engineered backfill soil. The resulting groundwater concentrations were calculated.

- The groundwater concentrations were compared to the LTP Groundwater DCGLs (Table 4-4) and a "future groundwater" dose was determined.
- The radioactivity content of any other remaining concrete structures in the former Radiological Control Area (RCA) was included in the containment basement calculation.
- Structures outside the RCA, such as the discharge tunnels and building/crane footings, were included in a separate calculation assuming all radioactivity from these structures travels into the discharge tunnels. As the discharge tunnels were not backfilled, the calculation assumes only groundwater is present.

CY utilized generally the same approach as Maine Yankee (See Section 5.3.2.3.2) except as follows:

- Although the steel liner in the In-Core Instrumentation (ICI) Sump at CY was to remain, no credit was taken in the dose model for the barrier to the leaching of radionuclides that it would provide. Maine Yankee did take credit for this barrier and assumed that the liner would remain intact for 50 years.
- In order to ensure that the ICI Sump would be considered inaccessible, the sump areas were filled with grout covering the area of activated concrete by at least 3 feet (0.9 m).
- Maine Yankee performed separate calculations for all of the various basements to remain on site. CY, as discussed above, assumed that all the activity would migrate to either the containment basement or to the discharge tunnels.

The following (Table 5-3) is a comparison of the DCGLs CY originally planned to use assuming the "Concrete Debris Scenario" and the higher concentrations which were actually allowed in any area of the CY site using the "Basement Fill Model (BFM)". As can be seen in Table 5-3, substantially higher concentrations, in one case 792 times higher, were allowed under the Basement Fill Model. Being able to leave behind concrete with these higher concentrations allowed CY to avoid removing the concrete from the Containment ICI Sump. It also avoided the projected need to completely remove the Spent Fuel Pool without the Basement Fill Model. The remediation needed behind the Fuel Pool liner was approximately 12 inches (30 cm) in the floor and essentially no remediation was needed in the walls. As the remediation of the Containment ICI Sump and the removal of the Spent Fuel Pool both would have been difficult remediation efforts, significant cost and schedule time were saved.

Table 5-3
Comparison of Default to Realistic CY Buried Concrete Concentrations

Key Radionuclide	Original CY "Default" DCGLs pCi/g (Bq/g)	Actual Concentrations Allowed using Basement Fill Model (BFM) pCi/g (Bq/g)	Ratio of BFM to Default
H-3	90.5 (3.35)	9,620 (356)	106:1
Fe-55	90 (3.33)	1,950 (72.2)	22:1
Co-60	2.7 (0.1)	2,140 (79.2)	792:1
Cs-137	13.5 (0.5)	311 (11.5)	23:1
Eu-152	227 (8.4)	20,100 (744)	89:1

5.3.2.2 Trojan

5.3.2.2.1 Derived Concentration Guidance Levels for Structures

The Trojan plant submitted its dose based DCGLs shortly after the License Termination Rule became effective in 1996. As the guidance and experience was very limited at that time (this was approximately 3 years before the development of DCGLs by other major decommissioning projects such as CY), Trojan utilized the NRC screening values published in Federal Register. This decision allowed Trojan to proceed with their decommissioning in a timely manner even though these DCGLs were very conservative.

The NRC screening values were determined based on very conservative assumptions intended to bound most conditions expected to be found at reactor sites. For ease of review by the NRC, Trojan provided D & D computer code runs using the conservative default parameters to provide the basis of the generic screening DCGLs.

5.3.2.2.2 Embedded Piping

EPRI Report "Remediation of Embedded Piping, Trojan Nuclear Plant Decommissioning Experience" (EPRI Report 1000908 Reference 5-4), provides a detailed discussion of experiences with embedded piping at the Trojan Plant. The following are the highlights of that report.

The Trojan Plant was designed with much more embedded piping than the other plants that have gone through decommissioning to date. It was estimated that there was 29,000 feet (5.5 miles or 8,839 meters) of embedded piping at Trojan. This piping was part of drain systems, embedded ventilation ductwork, buried embedded piping, and embedded conduit. Although much of the piping was short sections of piping systems passing through walls [generally 4 ft (1.2 m) in length], removal would be difficult and expensive.

Under Trojan's plans to release the site with buildings standing, it was more economical to leave this piping in place if it was shown to meet the limits described below. Reference 5-4 states that for a plant that is planning on removing the concrete that encases the embedded piping, it is likely more cost effective to grout the embedded piping and dispose of it with the remainder of the concrete in lieu of decontaminating and surveying the piping.

Dose Modeling Approach for Embedded Piping

Trojan decided to allot a portion of the 25 mrem/yr (0.25 mSv/yr) unrestricted release criteria to account for the presence of embedded piping. Trojan performed the following dose calculation based on allotting no more than 5 mrem/yr for embedded piping:

- All contamination inside the piping would be encapsulated by grouting the pipe. This would eliminate inhalation of airborne contamination as a dose pathway
- No more than 100,000 dpm/100 cm^2 (167,000 Bq/m^2) of beta/gamma radionuclide contamination would be allowed on the inside of any embedded piping to remain.
- To address the reduction of the allowable dose for the remainder of building surfaces, the NRC screening values (Building Occupancy Scenario) would be reduced by 20% and used for the Final Status Survey of the buildings at Trojan.

This methodology was approved by the NRC.

5.3.2.3 Maine Yankee

5.3.2.3.1 Contaminated Concrete

Maine Yankee (MY) developed two different gross beta DCGLs for two areas of the plant. A Special Area DCGL was designated for the containment pipe trench area where the presence of alpha radionuclides required the gross beta DCGLs to be scaled down to account for the alpha radionuclides. All other basement concrete [other than the Incore Instrumentation (ICI) Sump, which was assessed using the Basement Fill Model discussed in the next section] was to be surveyed to a higher limit. These limits could not be higher than the generic NRC surface screening values. The gross beta DCGL is determined by using the radionuclide fractions determined for these areas from characterization data. Using these preset surface contamination limits, the dose from the drinking water pathway due to leaching of the concrete contamination into the containment backfill, and subsequently groundwater, was calculated.

The soil to be used for backfill at Maine Yankee was tested by Brookhaven National Lab to determine its affinity for radionuclides (i.e., its distribution coefficient, K_d). These K_ds were used in the calculation of the equilibrium between the radioactivity leached into the groundwater and the soil used to backfill the building basements. These DCGLs were only used in basements as Maine Yankee decided to remove all above grade contaminated concrete for disposal as radioactive waste.

5.3.2.3.2 Basement Activated Concrete in Containment Incore Instrumentation Sump

Maine Yankee initially set the DCGL for basement activated concrete at 1 pCi/g (0.037 Bq/g.) As a result of characterization sampling, it was determined that the 3/8" carbon steel liner and up to 18 inches (0.5 m) of concrete would need to be removed from the In-Core Instrumentation (ICI) Sump surfaces (located below the Reactor Vessel) to achieve this DCGL. The original MY LTP containing to this DCGL was very conservative as it assumed no credit for the isolation that the liner provided and assumed that the entire radioactivity in the concrete would be released instantaneously into the backfill and groundwater. In a proposed amendment to the LTP (Reference 4-15, subsequently approved by NRC), Maine Yankee changed its approach to concrete remediation to one based on a more realistic dose modeling scenario.

In the new remediation approach, MY proposed to remove all of the concrete inside of the containment liner. This change would result in the removal of a larger volume of contaminated concrete but would leave a higher inventory of activated concrete behind. Even with this change, over 93 % of the activated concrete would be removed. MY proposed to leave concrete with concentrations well above the 1 pCi/g [3.7 E (-02) Bq/g] DCGL in the ICI Sump area. This change would result in a net increase in the total activity contained in activated concrete to be left on site.

In this more realistic dose scenario proposed, MY took credit for the barrier that the containment liner provided against leaching. MY applied the radionuclide concentrations measured in the rebar of the ICI sump as the radioactivity content of the liner. They assumed that the liner would corrode away over a period of 50 years. This was a conservative assumption because the actual projection for corrosion was 168 to 200 years. The liner is assumed to preclude the release of radionuclides from the concrete over the 50 years. MY also used diffusion rates from literature instead of the extremely conservative, instantaneous releases assumption used in the original LTP.

Maine Yankee also safety related justifications for not remediating the concrete and the steel liner. Maine Yankee described the required remediation as similar to a mine shaft installation being constructed 60 feet (18.3 m) below the surface. This was MY primary justification for not removing the concrete and steel liner as this remediation was projected to cost $800,000 would involve the following safety hazards:

- Fall hazard for personnel accessing and egressing the remediation location
- High temperature slag generated and concrete explosions during liner cutting
- Work would need to be conducted in a confined space (potentially oxygen deficient area)
- Smoke would be generated from the machinery used during the remediation
- Dangers introduced by the use of thermal cutting of rebar in a relatively small space

Maine Yankee stated that explosives would be needed to separate the liner from the concrete and that additional blasting would be needed to remove additional thickness of concrete. This blasting could introduce cracks in the surrounding concrete which could allow groundwater to enter the containment. This would be a reduction in the protection that the 7 feet (2.13 m) of concrete around the ICI Sump provided.

Maine Yankee calculated that the dose that would result from leaving the activated concrete and liner in place at 0.041 mrem/yr [4.1 E (-04) mSv/yr] and that based on such a small dose, removal of this material wasn't consistent with the ALARA principal. This approach was approved by the NRC.

5.3.2.3.3 Embedded Piping

The dose due to contamination on embedded piping was calculated assuming a uniform concentration on all the embedded piping for two categories of areas of Maine Yankee (As described below). The total inventory of this contamination was assumed to instantaneously release into the worst case building basement location. The released inventory would then mix with the backfill material and groundwater. The primary dose pathway is from drinking water. Different contamination limits for the Spray Pump Building Embedded Piping and for the remaining Balance of Plant (BOP) Embedded Piping were used due to the different radionuclide ratios in the two areas (See Table 4-9). The doses due to these two categories are given in Table 4-10.

5.3.2.4 Yankee Rowe

5.3.2.4.1 Derived Concentration Guidance Levels for Structures

The decommissioning approach at Yankee Rowe was to:

- Remove all structures to 4 feet (1.2 m) below final grade
- Use building concrete debris as fill
- Use 3 feet (0.9 meter) of clean cover to grade

The Concrete DCGLs for Yankee Rowe LTP were determined for the different media as follows:

- RESRAD and the Resident Farmer Scenario for concrete debris used as backfill material
- RESRAD Build and the Building Occupancy Scenario for:
  - Remaining Minor Buildings
  - Remaining Building Pads

The use of the RESRAD family of computer codes to calculate DCGL was very similar to the approach used at CY.

Additionally, for certain partial subsurface structures, Yankee Rowe utilized the Basement Fill Model in the following manner:

- All subsurface structures to remain were combined to represent a wall of average thickness
- H-3 assumed uniform throughout the concrete at 1 pCi/g (0.037 Bq/g)
- Sr-90, Co-60 and Cs-137 assumed uniform at 1 inch (2.54 cm) contamination depth
- Radionuclides are assumed to be released from concrete per diffusion rates listed in published studies
- The maximum activity released in 1 year was calculated
- The maximum activity calculated in the last bullet was distributed over the volume pumped from a postulated well of a future resident farmer over the course of 1 year
- Groundwater DCGLs calculated by RESRAD were used to calculated yearly dose from groundwater concentration pumped from the well
- Projected dose from all the partial subsurface structures to remain at Yankee Rowe was calculated as 0.5 mrem (0.005 mSv)

5.3.2.4.2 Adjusting Derived Concentration Guidance Levels to State of Massachusetts Site Release Criteria

The final step for the dose based limits for all media to adjust the 25 mrem/yr (0.25 mSv/yr) criteria to the State of Massachusetts Department of Public Health (DPH) 10 mrem/yr (0.10 mSv/yr) criteria. The Yankee Rowe Building Surface DCGLs corresponds directly to the 10 mrem/yr (0.10 mSv/yr) limit since the Building Occupancy Scenario is not affected by groundwater. Table 5-4 lists the resulting DCGLs.

Table 5-4
Yankee Rowe Derived Concentration Guidance Levels (Based on Meeting State of Massachusetts Department of Public Health 10 mrem/yr limit)

Radionuclide	Building Surface DCGL (Corresponds to 10 mrem/yr [0.1 mSv/yr]) dpm/100 cm2 (Bq/m2)
H-3	1.4 E+08 (2.3 E+08 Bq/m^2)
C-14	4.0 E+06 (7.3 E+06 Bq/m^2)
Fe-55	1.6 E+07 (2.7 E+06 Bq/m^2)
Co-60	7.2 E+03 (1.2 E+04 Bq/m^2)
Ni-63	1.5 E+07 (2.5 E+07 Bq/m^2)
Sr-90	5.6 E+04 (9.3 E+04 Bq/m^2)
Nb-94	1.0 E+04 (1.7 E+04 Bq/m^2)
Tc-99	5.6 E+06 (9.3 E+06 Bq/m^2)
Ag-108m	1.0 E+04 (1.7 +04 Bq/m^2)
Sb-125	4.0 E+04 (6.7 E+04 Bm^2)
Cs-134	1.2 E+04 (2.9 E+04 Bq/m^2)
Cs-137	2.5 E+04 (4.2 E+04 Bq/m^2)
Eu-152	1.5 E+04 (2.5 E+04 Bq/m^2)
Eu-154	1.4 E+04 (2.3 E+04 Bq/m^2)
Eu-155	2.6 E+05 (4.3 E+05 Bq/m^2)
Pu-238	2.3 E+03 (3.8 E+03 Bq/m^2)
Pu-239/240	2.0 E+03 (3.3 E+03 Bq/m^2)
Pu-241	1.0 E+05 (1.7 E+05 Bq/m^2)
Am-241	2.0 E+03 (3.3 E+03 Bq/m^2)
Cm-243/244	2.9 E+03 (4.8 E+03 Bq/m^2)

As the Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM) classification of a Survey Area was based on the potential for residual radioactivity and not the actual concentrations, the 25 mrem/yr (0.25 mSv/yr) criteria was used in classification of survey areas.

Although the NRC and State of Massachusetts DPH allowed a dose based criteria for concrete building debris , Yankee Rowe learned late in the decommissioning that they also needed to meet the more restrictive criteria required by another State of Massachusetts regulatory agency, the Department of Environmental Protection (MADEP). These criteria applied to any material used as backfill, such as concrete debris, asphalt and soil, through a Beneficial Use Determination (BUD). The limits approved through the BUD are given in Section 6.3.2 of the EPRI Report on "Concrete Characterization and Dose Modeling During Plant Decommissioning" (EPRI Report 1015502, Reference

1-3). As the MADEP limits are essentially a “no detectable activity” criteria, the Concrete Debris Scenario DCGLs previously discussed could not be used by Yankee Rowe.

5.3.2.5 Rancho Seco

As discussed in Chapter 3, the municipal utility owners of the Rancho Seco site intend to retain ownership of the site. This allows restrictions to be placed on the future use of the site and less conservative assumptions could be utilized in dose modeling. This restriction allows the average member of the critical group to be 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 is made that occupancy for all areas of the site would be limited to a 50 week year (45 hours per week). Additionally, Rancho Seco justified lower occupancy rates (as discussed below) for the more highly contaminated area of the site buildings. This reduced the amount of required remediation and resulted in less radioactive waste being generated.

The primary factor in the dose modeling for concrete at Rancho Seco is the fact that the buildings were left standing after license termination. When left standing,\ certain areas of the plant, and the concrete in the areas, can be considered inaccessible to workers. The best example of this is the upper portion of the liner in the containment building. As this area will not be accessible without a lift or the construction of scaffolding, different assumptions were used in the DCGL calculations than were used at other utilities.

Critical Group and Dose Pathways 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 (Reference 5-3). 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 considers 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 tritiated water vapor,
- Inadvertent ingestion of radioactive material directly from the source, and
- Ingestion of materials deposited on the surfaces of the building compartments.

Rancho Seco evaluated two scenarios in determining the Building Surface DCGLs that would apply inside of the containment building. These scenarios are described in the following sections.

5.3.2.5.1 Building Renovation/Demolition Scenario

The building renovation/demolition scenario, as described in NUREG/CR-5512, Vol. 1 (Reference 5-3) along with the input data template and input parameter values provided in ANL/EAD/03-1 (Reference 5-5), specify the use of a volume source with a thickness of 15 cm (5.9 inches). In the case of the containment building, any residual contamination was likely to 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 5-5 were determined using the RESRAD-Build code. The largest factor in this scenario is the occupancy time. The occupancy time was taken to be 63 days versus the standard value of 200 days for the unrestricted building occupancy scenario, according to NRC guidance.

5.3.2.5.2 Industrial Worker Scenario

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

Table 5-5 lists the DCGLs determined for these two scenarios along with those using the standard assumptions for Building Occupancy without restrictions (as was done at a number of plant decommissioning 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. This methodology was expected to result in lower costs due to a facilitated Final Status Survey and less building remediation.

Table 5-5
Comparison of Rancho Seco Building Surface Derived Concentration Guidance Levels for Alternate Scenarios

Radio-nuclide	Renovation/Demolition Scenario - dpm/100 cm^2	Industrial Worker Scenario DCGL - dpm/100 cm^2	Unrestricted Access 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

Notes 1: These radionuclides were not detected in significant quantities in Rancho Seco soil samples. As allowed by NRC guidance, they were excluded from further consideration during the FSS. 2. 1 dpm/100 cm^2 = 1.67 Bq/m^2

5.3.2.5.3 Embedded Piping

The embedded piping scenario used at Rancho Seco assumes that the piping remains in place following decommissioning. It is assumed that the dose to the industrial worker is from direct gamma exposure from the residual activity in the pipe. An allowance is made for photon attenuation by the wall or floor thickness of concrete remaining over the pipe in this dose calculation. The dose from the embedded piping is added to the dose from the residual activity on the walls or floors of the room in which the embedded piping is present. 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.

5.3.2.5.4 Bulk Materials

Although Rancho Seco considered the building surface DCGLs 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 single nuclide DCGLs for bulk materials in order to evaluate these surfaces and the underlying volumetric contamination 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 would have a potential of being activated and require a Final Status Survey. 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 (164 yd^2) derived for the Building Occupancy DCGLs was used by replacing the floor surface source with a 1 foot (0.3 m) thick (the most likely maximum depth of activation or contamination according to NUREG/CR-5884, Volume 2 (Reference 6-1)) 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 6-2 lists the Bulk Material DCGLs for Rancho Seco.

Table 5-6
Rancho Seco Bulk Material Derived Concentration Guidance Levels

Radio-nuclide	DCGL pCi/g (Bq/g)	Radionuclide	DCGL pCi/g (Bq/g)
H-3	7.86 E+03 (291)	Cs-137	3.38 E+01 (1.25)
C-14	1.60 E+06 (5.92 E+03)	Pm-147	1.64 E+06 (6.07 E+04)
Na-22	8.39 E+00 (0.31)	Eu-152	1.64 E+01 (0.61)
Fe-55	3.91 E+07 (1.45 E+06)	Eu-154	1.50 E+01 (0.56)
Ni-59	1.49 E+07 (5.51 E+05)	Eu-155	7.81 E+02 (28.9)
Co-60	7.06 E+00 (0.26)	Np-237	7.49 E+01 (2.77)
Ni-63	6.85 E+06 (2.53 E+05)	Pu-238	3.61 E+02 (13.4)
Sr-90	4.16 E+03 (154)	Pu-239	1.23 E+02 (4.55)
Nb-94	1.18 E+01 (0.44)	Pu-240	2.96 E+02 (11)
Tc-99	7.37 E+05 (2.73 E+04)	Pu-241	2.05 E+04 (759)
Ag-108m	1.20 E+01 (0.44)	Am-241	2.70 E+02 (10)
Sb-125	4.75 E+01 (1.76)	Pu-242	3.09 E+02 (11.4)
Cs-134	1.22 E+01 (0.45)	Cm-244	6.72 E+02 (24.9)

5.3.2.6 San Onofre Unit 1(SONGS-1)

The SONGS-1 plant has proceeded with much of its decommissioning without having submitted an LTP to the NRC. As previously discussed, SONGS-1 needs to remove all the structures to at least 1 meter below grade per its agreement with the site owners (a branch of the United States military). To date, any concrete debris created by the decommissioning or concrete that could not be left in place (because it could not be "free released" using the sites operating plant procedures) has been disposed of as radioactive waste.

Section 6: Site Release Limit Development for Buried Piping

6.1 Overview

NRC guidance has primarily discussed site release limit development for land areas and structures. There are other media which are not included in the guidance and need to be addressed during the site release limit approval process so that numerical limits are available when the Final Status Surveys are being conducted. Sediments, embedded piping and bulk materials are typically evaluated with land areas and building occupancy scenarios so they have been included in Chapters 4 and 5. This chapter addresses experiences with the development of site release limits for buried piping.

6.2 Buried Piping Dose Modeling

6.2.1 Connecticut Yankee Experience

The radioactivity associated to buried piping in contact with the saturated zone at Connecticut Yankee (CY) was analyzed to determine surface activity limits that would result in no more than a 1 mrem/yr (0.01 mSv/yr) dose. If the dose from buried piping could be shown to be no more than 1 mrem/yr, the dose could be considered insignificant and not require inclusion in the Equation 4-1, the CY Compliance Equation. Part of the dose model for this type of piping was that the pipe would be grouted after any required remediation and surveying. To simplify the analysis, the piping material was assumed to be eroded away, leaving the slug of grout with the contamination from the interior surface of the piping. Consistent with these simplified assumptions, the Derived Concentration Guidance Levels (DCGLs) calculated for the Concrete Debris Scenario (Table 5-2) were applied to determine the dose from the slug of grout.

In order to calculate the release limits for the piping [corresponding to 1 mrem/yr (0.01 mSv/yr)], first, the portion of the 25 mrem/yr (0.25 mSv/yr) dose from concrete debris due to water dependent pathways was determined for each radionuclide. The concentrations for each radionuclide corresponding to the water dependent dose (a fraction of the total dose from all pathways for concrete debris) were then ratioed to represent a concentration (volumetric contamination limit) that would result in 1 mrem/yr (0.01 mSv/yr.) (i.e. using the ratio volumetric limit/25 mrem/yr = normalized volumetric limit at 1 mrem/yr)

Finally, the volumetric contamination value was converted to surface contamination levels using various diameters of piping. The release limits to be applied to this piping are given in Table 6-1 for a 4 inch (10.16 cm) pipe. Limits for pipes of different diameters were determined by applying the same methodology.

Table 6-1
Release Limits for Buried Piping (For a 4 inch (10 cm) diameter pipe)

Radionuclide	Surface Limit Resulting in 1 mrem/yr (0.01 mSv/yr) Dose (dpm/100cm^2)
H-3	5.21 E+03
C-14	7.77 E+04
Mn-54	5.31 E+04
Fe-55	6.17 E+04
Co-60	3.21 E+05
Ni-63	1.52 E+05
Sr-90	1.87 E+02
Nb-94	1.37 E+05
Tc-99	2.44 E+04
Ag-108m	1.37 E+06
Cs-134	8.35 E+04
Cs-137	9.66 E+04
Eu-152	2.68 E+05
Eu-154	1.87 E+05
Eu-155	1.20 E+06
Pu-238	7.50 E+02
Pu-239	6.82 E+02
Pu-241	1.14 E+04
Am-241	3.33 E+02
Cm-243	4.61 E+02

Note: 1 dpm/100 cm^2 = 1.67 Bq/1m^2

6.2.2 Maine Yankee Experience

6.2.2.1 Buried Piping, Conduit and Cable

The dose model for buried piping at Maine Yankee assumes that the total amount of buried piping is contaminated at the levels given in Table 4-9, "*Maine Yankee Contaminated Material DCGLs*". The piping is assumed to degrade and the contamination distributed through a soil volume corresponding to the volume of the pipe. The resulting concentration is taken to be in the deep soil. Therefore, the dose due to the buried piping is calculated as if it were deep soil with 3 feet (1 meter) of cover material. All piping left on site was located at least 3 feet below the surface. Table 4-10 lists the calculated dose.

6.2.3 Rancho Seco Experience

6.2.3.1 Buried Piping

The buried piping scenario used by Rancho Seco incorporates the soil DCGL values discussed in Section 4.3.2.1.2. 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. It was assumed that the disintegrated media was contaminated to soil DCGL concentrations obtained 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 typical 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.

Section 7: Summary and Conclusions

The purpose of this document was to present the experiences in the United States and Spain concerning the development of numerical limits for residual radioactivity in soil, structures, groundwater and other media to be used for the unconditional release of nuclear power plant sites. The experiences and regulatory approved approaches documented span from 1996 to the present. This timeframe covers the period when dose based standards have been used by the U.S. Nuclear Regulatory Commission (NRC) to release power plant sites.

A number of power plant sites have achieved release of their sites from NRC license using the site future use assumptions and dose modeling scenarios discussed in this document.

Guidance issued at the same time as the NRC License Termination Rule in 1996 led some power plant licensees to choose conservative dose modeling scenarios that resulted in conservative release limits. More recent NRC guidance, discussed in this report, allows the use of more realistic future site use assumptions and dose modeling scenarios based on those more realistic assumptions. Use of these alternate scenarios can result in higher site release concentration limits, less remediation, less radioactive waste and a reduction in cost while protecting the future user of the site.

Section 8: References

1-1 Code of Federal Regulations, Title 10, Part 20.1402, "Radiological Criteria for Unrestricted Use."

1-2 EPRI Report # 1015500. "Final Status Survey Experience Report", March 2008.

1-3 EPRI Report # 1015502. "Concrete Characterization and Dose Modeling During Plant Decommissioning: Detailed Experiences 1993 – 2007", March 2008.

1-4 EPRI Report # 1019228. "Characterization and Dose Modeling of Soil, Sediment and Bedrock During Plant Decommissioning", November 2009.

1-5 EPRI Report # 1021108. "Use of In-situ Gamma Spectroscopy During Nuclear Power Plant Decommissioning" December 2010.

2-1 NUREG 1757, "Consolidated Decommissioning Guidance", Volume 2, Rev. 1 Sept 2006

2-2 NUREG-1575/EPA 402-R-97-016, "Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)

2-3 NRC Policy Issue, SECY-03-0069, Results of the License Termination Rule Analysis, May 2, 2003.

2-4 NRC Regulatory Issue Summary 2004-08," Results of the License Termination Rule Analysis" May 28, 2004

3-1 Oak Ridge Associated Universities Website address, www.orau.gov/ddsc/dose/compcode.htm

3-2 Argonne National Laboratory Website address, www.ead.anl.gov/project

3-3 NRC Regulatory Guide 1.86, Termination of Operating Licenses for Nuclear Reactors, June 1974.

3-4 Federal Resister, Volume 64, No. 234, dated December 7, 1999, Generic Radionuclide Screening Values.

3-5 Big Rock Point Restoration Project License Termination Plan, Revision 0, April 1, 2003.

3-6 European Commission Document, Radiation Protection 113: Recommended Radiological Protection Criteria for the Clearance of Buildings and Building Rubble from the Dismantling of Nuclear Installations - Recommendations of the Group of Experts Set Up Under the Terms of Article 31 of the Euratom Treaty, January 2000

4-1 EPRI Report # 1006949, Use of Probabilistic Methods in Nuclear Power Plant Decommissioning Dose Analysis, May 2002

4-2 EPRI Report # TR-112874, "Comparison of Decommissioning Dose Modeling Codes for Nuclear Plant Use: RESRAD and D&D, November 1999

4-3 EPRI Report # 1015118, Groundwater Protection Guidelines for Nuclear Power Plants, November 2007

4-4 U.S. Nuclear Regulatory Commission, Memorandum of Understanding between the Environmental Protection Agency and the Nuclear Regulatory Commission, "Consultation and Finality on Decommissioning and Decontamination of Contaminated Sites", October 9, 2002. (ML073250093, ML022830208)

4-5 EPA Document # 816-D-00-002, Implementation Guidance for Radionuclides, January 2002, Appendix I has the MCLs

4-6 Haddam Neck Plant License Termination Plan, Revision 4, November 2006

4-7 U.S. Nuclear Regulatory Commission Guidance, NUREG/CR-6697, Development of Probabilistic RESRAD 6.0 and RESRAD-BUILD 3.0 Computer Codes, Dec 2000, (ML010090284).

4-8 Argonne National Laboratory, ANL/EAD-04, "User's Manual for RESRAD Version 6," July 2001.

4-9 Rancho Seco License Termination Plan, Chapter 6, Revision 0 dated April 2006.

4-10 International Commission on Radiological Protection, Principles of Monitoring for the Radiation Protection of the Population A Report of Committee 4 of the ICRP (adopted 1984), ICRP 43, 1985

4-11 Argonne National Laboratory, NUREG/CR-6697, "Development of Probabilistic RESRAD 6.0 and RESRAD-BUILD 3.0 Computer Codes", December 2000

4-12 U.S. Nuclear Regulatory Commission, NUREG/CR-6692, "Probabilistic Modules for the RESRAD and RESRAD-BUILD Computer Codes", November 2000

4-13 U.S. Nuclear Regulatory Commission, NUREG/CR-6676, "Probabilistic Dose Analysis Using Parameter Distributions Developed for RESRAD and RESRAD-BUILD Computer", May 2000.

4-14 IAEA Safety Standards Series No. RS-G-1.7, Application of the Concepts of Exclusion, Exemption and Clearance Safety Guide, September 2004.

4-15 Maine Yankee License Termination Plan, Revision 3, dated October 15, 2002

5-1 Federal Register, 63 FR 64132, Supplemental Information on the Implementation of the Final Rule on Radiological Criteria for License Termination, dated November 18, 1998.

5-2 Haddam Neck License Termination Plan, section 6.0, revision 2 draft.

5-3 U.S. Nuclear Regulatory Commission, NUREG/CR–5512, Volume 1, "Residual Radioactive Contamination From Decommissioning: Technical Basis for Translating Contamination Levels to Annual Total Effective Dose Equivalent," October 1999.

5-4 EPRI Report # 1000908, "Remediation of Embedded Piping, Trojan Nuclear Plant Decommissioning Experience", dated October 2000.

5-5 Argonne National Laboratory, ANL/EAD/03-01, "User's Manual for RESRAD-BUILD Version 3," June 2003.

6-1 U.S. Nuclear Regulatory Commission, NUREG/CR-5884, Volume 2, "Revised Analyses of Decommissioning for the Reference Pressurized Water Reactor Power Station," Draft Report for Comment, October 1993.

The Electric Power Research Institute Inc., (EPRI, www.epri.com) conducts research and development relating to the generation, delivery and use of electricity for the benefit of the public. An independent, nonprofit organization, EPRI brings together its scientists and engineers as well as experts from academia and industry to help address challenges in electricity, including reliability, efficiency, health, safety and the environment. EPRI also provides technology, policy and economic analyses to drive long-range research and development planning, and supports research in emerging technologies. EPRI's members represent more than 90 percent of the electricity generated and delivered in the United States, and international participation extends to 40 countries. EPRI's principal offices and laboratories are located in Palo Alto, Calif.; Charlotte, N.C.; Knoxville, Tenn.; and Lenox, Mass.

Together...Shaping the Future of Electricity

Programs:

Nuclear Power

Decommissioning and Technology Development

1025315

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