BRUCE MONTGOMERY
Director
Decommissioning and Used Fuel

1201 F Street, NW, Suite 1100
Washington, DC 20004
P: 202.406.0054
mab@nei.org
nei.org

NUCLEAR ENERGY INSTITUTE

April 11, 2023

Mr. James C. Corbett
Chief Financial Officer (Acting)
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Subject: Fee Exemption Request for Review and Endorsement of NEI 22-01, "License Termination Process"

Project Number: 689

Dear Mr Corbett:

By letter dated February 14, 2023, the Nuclear Energy Institute (NEI)[1] submitted technical report NEI 22-01, "License Termination Process," for review and endorsement (ML23045A322, ML23045A323, ML23045A324). NEI requested that NRC's review, including any subsequent submittals necessary to address staff review comments, be granted a fee waiver pursuant to 10 CFR 170.11 (ML23045A337).

During subsequent discussions with your staff, it was suggested that NEI revise the non-proprietary version of NEI 22-01 to acknowledge information that was available in the public domain. Accordingly, NEI has developed a revised version of the non-proprietary document for your use (attached).

If you have any questions on this matter, please contact me at bsm@nei.org.

Sincerely,

Bruce Montgomery

Cc:

Ms. Jane Marshall, NMSS/DUWP
Mr. Bruce Watson, NMSS/DUWP
Mr. Cynthia Barr, NMSS/RTAB
Mr. Shaun Anderson, NMSS/DUWP/RDB
NRC Document Control Desk

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[1] The Nuclear Energy Institute (NEI) is responsible for establishing unified policy on behalf of its members relating to matters affecting the nuclear energy industry, including the regulatory aspects of generic operational and technical issues. NEI's members include entities licensed to operate commercial nuclear power plants in the United States, nuclear plant designers, major architect and engineering firms, fuel cycle facilities, nuclear materials licensees, and other organizations involved in the nuclear energy industry.

NEI TECHNICAL REPORT

NEI 22-01, [Rev 0]
Non-Proprietary Document, Revision 1

License Termination Process

Prepared by the Nuclear Energy Institute
January 2023

Revision Table

Revision	Description of Changes	Date Modified	Responsible Person

<u>Acknowledgements</u>

This document was developed by the Nuclear Energy Institute. NEI acknowledges and appreciates the contributions of NEI members in providing input, reviewing and commenting on the document including:

NEI Project Lead: Bruce Montgomery

Industry Lead: Gerry Van Noordennen, Energy*Solutions*

<u>Industry Team:</u>
Bill Barley, Pacific Gas and Electric Company
Ron Cardarelli, C.N. Associates Inc.
Eric Darois, Radiation Safety & Control Services, Inc.
Jean Fleming, Holtec International
Willie Harris, C.N. Associates Inc.
Rich McGrath, EPRI
Sarah Roberts, Energy*Solutions*
Kris Rowberry, Pacific Gas and Electric Company

Executive Summary

The decommissioning of U.S. commercial nuclear reactors is evolving through both technical innovation and adaptation in business models. Accelerated decommissioning, or DECON, has become the preferred pathway for recently shut down reactors, and some owners of reactors previously placed in SAFSTOR are favorably viewing the advantages of shifting to a DECON strategy. Companies specializing in radiological decontamination and major dismantling projects have emerged as highly successful general contractors and even owner-licensees.

With this evolution in the marketplace, the license termination process is taking on greater importance in the successful completion of these projects. The industry's preferred endpoint for license termination is gaining NRC approval to release the plant site for unrestricted use so that it be repurposed by either the owner or by the public.

The process of achieving unrestricted release of a site is highly technical and complex, requiring skill sets on the part of both the licensee and the regulator that are in short supply. The recent surge in the number of reactors prematurely ceasing operation and going into DECON is challenging these resources, resulting in increased back-end timelines and escalating costs, even as the front-end of these projects are experiencing reduced timelines and increased cost-effectiveness.

The license termination process begins with the submittal of a License Termination Plan (LTP) by the licensee to the NRC. The LTP becomes the licensee's commitment for how the criteria for site remediation will be met and is an enforceable component of the license held by the owner.

A major aspect of any LTP is the licensee's plan for conducting final status surveys (FSS) of the site, along with the analyses that will be completed for these surveys to show that the objectives of the plan have been satisfied. Execution of the FSS and achieving NRC concurrence on the adequacy of both the surveys and the analytical results is a major undertaking and has been the source of significant project delays.

NRC has published and maintains a large compendium of both process and technical guidance on the conduct of decommissioning. Because NRC's decommissioning oversight responsibilities extend over a wide range of facility types and owners, both private and public, this guidance is necessarily voluminous and complex, and must also account for the unique needs of and the jurisdictional interfaces with the Environmental Protection Agency, the Department of Defense, and the Department of Energy. Requirements for the format and content of an LTP, and the level of detail required in FSS reports, for example, are defined at a high level to allow for diversity in the user community. For the typical commercial nuclear power reactor licensee, further definition in these requirements is warranted and is expected to be extremely useful to license termination applicants.

In addition, NRC's current guidance does not satisfactorily address a few technical subjects that have emerged as issues in ongoing license termination reviews. These include:

- How to develop and execute survey plans for subsurface soils and below grade structures,
- How to survey and account for discrete radioactive particles, and
- How to analytically deal with hard-to-detect radionuclides.

The industry believes that these issues can be addressed in a risk-informed manner that provides reasonable assurance of adequate protection of the public.

Finally, recent experience has highlighted the need for better coordination and communication between the licensee, the NRC region, and NMSS headquarters staff.

The objective of this report is to provide industry guidance on how to develop and write an LTP that includes a properly focused FSS plan for a typical commercial nuclear power reactor, including:

- Navigating the many regulatory documents that provide decommissioning guidance, and directing the users' attention to those requirements, practices and recommendations that are most applicable to the typical commercial reactor preparing for or undergoing decommissioning.

- Proposing a standard LTP format and content that can be used by both licensees and NRC reviewers to cross-reference against NRC's NUREG-1700 with guidance on an acceptable level of detail for each section.

- Providing a recommended communications protocol and checklist to follow in setting up interactions between stakeholders, including the NRC and state and local authorities throughout project planning and execution.

- Providing technical solutions for areas not currently addressed in NRC or EPA regulatory guidance.

- Providing guidance on the documentation of FSS findings and reports submitted to NRC for review.

By providing this technical document to the industry, NEI hopes to increase the efficiency of the license termination process by improving the quality of, consistency between, and adherence to license termination plans submitted to NRC for review. In doing so, the industry seeks to achieve NRC review and approval of LTPs on the order of one year, and completion of regulatory reviews of FSS reports in a timely fashion such that licenses can be terminated within one year of the completion of confirmatory radiological surveys at the site.

Table of Contents

1 INTRODUCTION TO THE LICENSE TERMINATION PROCESS

1.1 General Information

A goal for the License Termination Plan (LTP) should be to have the technical details for a Phase 1 Site Characterization well defined, Final Status Survey (FSS) capabilities developed, and site-specific release criteria (Derived Concentration Guideline Levels [DCGLs]) complete prior to initiation of decommissioning, particularly if early on-site excavations and backfill are anticipated. Whenever this is not possible, due to earlier than anticipated shutdown, ensure adequate information is obtained before any excavation backfill, including pedigree of backfill material (soil characteristics, radiological and hazardous chemical content), to demonstrate compliance with the end state clearance criteria.

Approval of the LTP should be planned prior to major site excavation of potentially contaminated land areas of the site and any partial site release of the licensed property. Time frames for NRC LTP approval could be 2 years or more from time of submittal, so plan appropriately. If demolition, remediation and backfill activities begin before NRC approval of the LTP, then site management must be aware that these activities are being conducted at risk.

1.1.1 Communications among Licensee and Regulators

Decommissioning licensees should begin discussions with NRC and state regulators, and other stakeholders, during the early stages of LTP development, with a focus on the issues that pose the highest risk to project completion and the approaches that will be taken to achieve remediation targets and successful release of the site for unrestricted use. Early agreement should be sought for the process and timeline for NRC review and approval of the LTP, including NRC approval of the licensee committed quality assurance (QA) and quality control (QC) programs, the plan for NRC inspections and independent NRC confirmatory surveys, and the resources that will be allocated to support regulatory reviews during license termination. The purpose of this document is to focus activities and reviews on the most risk-significant issues regarding public health and safety throughout the decommissioning and license termination process.

To facilitate NRC reviews, licensees should consider a phased approach to submitting LTP sections. For example, Chapter 2, “Site Characterization,” is a prime candidate for early submittal and NRC review/approval (See Appendices D & E). Following submittal of the LTP to NRC, a phased review of sections of the LTP should be considered and discussed with the NRC, and requests for additional information could be handled on a chapter-by-chapter basis on an agreed upon schedule.

During actual decommissioning, remediation, and survey activities, periodic meetings should be planned to discuss LTP progress, results of NRC Regional interface/inspections, MARSSIM implementation issues, etc. The periodicity of the meetings and attendance should be adjusted as needed to ensure adequate discourse between the licensee, NRC headquarters and NRC Regional personnel, and state regulatory agencies. A recommended protocol for interfacing with NRC and state regulators is provided in Appendix D, and a typical timeline for license termination-related activities and associated regulatory interactions is provided in Appendix E.

1.1.2 Standard Format and Content

Standard format and minimal content for the LTP is found in Regulatory Guide 1.179 (Reference 1). As with all Regulatory Guides, the prescribed format and content is pre-approved if followed. Any changes based on unique site characteristics or other issues will require additional justification.

1.1.3 Acceptance Criteria and Regulatory Review

The guidance to NRC LTP reviewers is contained in NUREG-1700. Author(s) of individual chapters of the LTP should familiarize themselves with the acceptance criteria, along with the Reg Guide 1.179 guidance, prior to each chapter's development. Where previous LTP submittals for other facilities are used as a guide, caution must be taken to ensure differences in site characteristics are accounted for.

1.1.4 Crosswalk between LTP and NUREG-1700

Appendix C provides a template for a crosswalk between the LTP and R.G. 1.179. Use of this template and including it in the LTP submittal will facilitate regulatory reviews.

1.1.5 History of Partial Site Releases

Licensees can release a portion of their licensed site for unrestricted use prior to receiving approval of a license termination plan per 10 CFR 50.83.

1.1.5.1 Partial Site Release Requirements

Nuclear power reactor licensees need to include the following information in their written request for NRC approval of the partial site release:

1) Evaluate the effect of releasing the property to ensure that:

    i. The dose to individual members of the public does not exceed the limits and standards of 10 CFR Part 20, Subpart D.

    ii. There is no reduction in the effectiveness of emergency planning or physical security.

    iii. Effluent releases remain within license conditions.

    iv. The environmental monitoring program and offsite dose calculation manual are revised to account for the changes.

    v. The siting criteria of 10 CFR Part 100 continue to be met.

    vi. All other applicable statutory and regulatory requirements continue to be met.

2) Perform a historical site assessment of the part of the facility or site to be released; and

3) Perform surveys adequate to demonstrate compliance with the radiological criteria for unrestricted use specified in 10 CFR 20.1402 for impacted areas.

[redacted]

1) The information specified above.

2) A reason why the impacted area needs to be removed from the license before the LTP is approved

3) The methods used and results obtained from the radiation surveys required to demonstrate compliance with the radiological criteria for unrestricted use specified in 10 CFR 20.1402.

4) A supplement to the environmental report, under § 51.53, describing any new information or significant environmental change associated with the licensee's proposed release of the property.

The NRC notices receipt of the release approval request or license amendment application and makes the approval request or license amendment application available for public comment. Before acting on an approval request or license amendment application submitted in accordance with 10 CFR 50.83, the NRC conducts a public meeting in the vicinity of the licensee's facility for the purpose of obtaining public comments on the proposed release of part of the facility or site. The NRC publishes the document in the Federal Register and in a forum, such as local newspapers, which is readily accessible to individuals in the vicinity of the site, announcing the date, time, and location of the meeting, along with a brief description of the purpose of the meeting.

1.1.5.2 Partial Site Release Process

[redacted]

1.1.5.2.1 Phase 1 – Redefining the Licensed Site Boundary

[redacted]

[redacted]

- [redacted]

[redacted]

It is important to note that 10 CFR 50.83(a) allows for the partial site release of non-impacted and impacted areas, under certain conditions. A non-impacted area is removed from the NRC license by an NRC approval letter. A Part 50 or Part 72 license may contain a license condition or a technical specification describing the licensed site area in detail. [redacted]

1.1.5.2.2 Phase 2 - Preparation of Program Documents

[redacted]

- [redacted]

- [redacted]

  [redacted]

- [redacted]

  [redacted]

- [redacted]

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  [redacted]

- [redacted]

  [redacted]

These documents will be prepared using guidance from MARSSIM and NUREG-1757, “Consolidated NMSS Decommissioning Guidance – Characterization, Survey, and Determination of Radiological Criteria, Volume 2.” [redacted]

It should be noted that this NEI guidance document contains many references to NUREG-1757, Volume 2. When reference is made to this NUREG, the reference is to Revision 2 of Volume 2 which was issued in July of 2022 unless otherwise noted. In some cases, the reference is made to NUREG-1757, Volume 2 Revision 1 when the source of the information in this NEI guidance uses Revision 1 as a reference.

1.1.5.2.3 Phase 3 – Survey Implementation for Partial Site Release

[redacted]

[redacted]

[redacted] Survey design will be performed in accordance with MARSSIM using the DQO process. The survey will utilize a graded approach with scan and sampling frequencies based on the potential for contamination concentrations in excess of the release criteria. [redacted]

[redacted]

1.1.6 Process for LTP revisions

NUREG-1700, Appendix B, "LTP Areas That Cannot Be Changed Without NRC Approval," presents specific instances that require NRC approval before implementing these types of changes to the LTP. Once the LTP is approved, these requirements will be integrated into the NRC license. Any changes that are made to the approved LTP that do not impact a license condition are allowed without prior NRC approval. A summary of changes made is required to be submitted with the biennial update to the FSAR. A process should be in place to control changes while the NRC is reviewing license amendments.

Examples of areas that may require prior NRC approval are changes that:

- Require Commission approval pursuant to 10 CFR 50.59.
- Result in significant environmental impacts not previously reviewed.
- Detract or negate the reasonable assurance that adequate funds will be available for decommissioning

- Decrease a survey unit area classification (e.g., impacted to not impacted, Class 1 to Class 2; Class 2 to Class 3; or Class 1 to Class 3 without providing NRC a minimum 14-day notification prior to implementing the change in classification.)

- Increase the derived concentration guideline levels and related minimum detectable concentrations (for both scan and fixed measurement methods).

- Increase the radioactivity level, relative to the applicable derived concentration guideline level, at which an investigation occurs.

- Change the statistical test applied to one other than the Sign test.

- Increase the Type I decision error.

2 SITE CHARACTERIZATION

The regulation applicable to this area of review is 10 CFR 50.82(a)(9)(ii)(A).

Regulatory Guide 1.179 describes the purpose of site characterization to "ensure that the licensee conducts final radiation surveys in all areas where contamination existed, remains, or has the potential to exist or remain." Guidance for designing and performing site characterization is provided in NUREG-1575 (MARSSIM) and NUREG-1757. Note characterization is performed for both radiological and non-radiological contaminants. The radiological data provides important input to Chapters 5 and 6, whereas the non-radiological data is described as part of Chapter 8.

Regulatory guidance allows the licensee to submit the entire site characterization package separately at any time before submitting the LTP, or as part of Chapter 2 of the LTP.

Chapter 2 also typically includes a summary of the Historical Site Assessment (HSA) – which is a precursor to site characterization.

Robust and defensible characterization data are integral to developing the final status survey methodology (Chapter 5) and site-specific DCGLs for the dose modeling assumptions (Chapter 6). Information presented in Chapter 2 includes (but is not limited to) radiological contamination incidents, locations of impacted and non-impacted portions of the site, characterization survey design including Data Quality Objectives (DQOs), instrumentation and the Data Quality Assessments (DQAs) performed. The characterization data includes survey results to include a discussion on background studies performed, potential ROCs, impacted structures and systems, non-impacted land, impacted land, surface and ground water, areas that were inaccessible during the initial site characterization, etc.

An individual qualified and experienced in site characterization typically performs characterization. Often the radiological and non-radiological characterizations are two separate and distinct areas of expertise.

As stated in NUREG 1.179, "the LTP site characterization should be sufficiently detailed to allow the NRC to determine the extent and range of radiological contamination of structures, systems (including sewer systems, waste plumbing systems, floor drains, ventilation ducts, and piping and embedded piping), rubble, and paved parking lots (both on and beneath the site). It should also include data on ground water, surface water, components, residues, and the environment, as well as the maximum and average contamination levels." Accordingly, the technical defensibility of the site characterization cannot be overstated.

2.1 Objectives of Site Characterization

It is important to recognize that site characterization is an iterative process and is not necessarily complete with one characterization campaign. First, characterization for purposes of waste disposal or occupational radiation exposure may differ substantially for purposes of identifying the abundance of radionuclides to support DCGL development or for FSS planning purposes. The differing purposes may lead to different requirements for minimum detectable concentrations (MDCs), quality assurance needs, sample volume and preservation methods, sample collection methods and more. For the remainder of this document, we assume that the purpose of characterization is to satisfy the development of DCGLs, the final list of radionuclides of concern (ROCs), and to support final status survey (FSS) planning including sample collection and analysis methods.

The ultimate objective in obtaining this characterization information should be to ensure that the list of radionuclides used to determine the DCGLs, and ROCs is complete. Otherwise, discovery of data gaps in this area could require an LTP change with NRC review and approval, thereby potentially creating schedule and cost impacts. One of the considerations in collecting this type of characterization data is to ensure that enough radionuclide data is generated to perform meaningful statistical analysis with consideration given to the potential that different buildings or rooms within a plant may exhibit very different radionuclide profiles. Therefore, enough data should be generated to evaluate the statistical variance at local area levels. Although it is possible that some areas may not be reasonably accessible for the collection of characterization samples due to radiological conditions, this potential should be well understood. This could include areas that may have been subjected to neutron activation as well as contamination from airborne release or liquid spill events throughout the plant's history. The sections below provide some consideration for the types and numbers of samples as well as the analysis strategies.

Lastly, some consideration should be given to the presence of discrete radioactive particles (DRPs) through information contained in the historical site assessment (HSA) and/or continued characterization. The presence of DRPs will depend largely on historical events, the methods used to manage (and size-reduce) high activity source terms (such as reactor vessel internal segmentation) and the movements/transfers of decommissioning wastes during the D&D process. If the likelihood of DRPs being present is low, then additional DRP surveys may be avoided.

If needed, the detection of DRPs will require the use of special survey scanning techniques and equipment. The determination of the sensitivity (i.e., MDAs) for DRP surveys can be complex because the detector geometry continuously changes during the radiation scanning process. Also, the potential dose to people from DRPs can be a complex problem involving the external and internal dose pathways to hypothetical exposure scenarios. Additionally, there is no regulatory standard that would apply to hypothetical doses from the presence of DRPs, therefore, any calculated doses cannot be measured against a compliance standard such as is done for the presence of more homogeneous residual radioactivity using the guidance in MARSSIM (Reference 2).

Appendix H provides guidance on DRPs (future).

Additionally, site characterization is needed to:

- Classify survey areas in accordance with MARSSIM
- Determine the degree of remediation needed to achieve the site release limits (DCGLs)

2.2 Types and Numbers of Samples

The types of samples can include concrete volumetric cores or dust, surface smears, metal cuttings (such as the containment liner for a PWR), soil and sediment. However, for concrete, experience has shown that contaminants can be found greater than 6 inches deep depending on the history of the area, and the radionuclide profile can change with depth, particularly for H-3, C-14, and Cs-134 from neutron activation of Cs-133 (from the decay of Xe-133) in certain areas. Surface smears may provide data on the presence of certain radionuclides, but they provide no information on depth. For soils and sediment, generally, the concentrations are low so the ability to identify and quantify the presence of the lower abundance radionuclides may be compromised.

We have identified four characterization types for concrete (and in some cases, other materials): traditional core-bores, collection of concrete dust through small hollow-core concrete drilling, laser ablation of concrete surfaces, and *in-situ* gamma spectroscopy. Each of these can quantify radionuclide profiles depending on the laboratory analysis strategy, but the last can only quantify gamma-emitting radionuclides and may not perform well in elevated radiation fields. Each of these is further discussed in the subsections below. One last technique not discussed or described here is by simple mechanical means of removing concrete with a hand-operated power tool or a hammer and chisel, both of which can collect concrete but in a less-controlled fashion regarding depth and surface area, as compared to other methods.

In evaluating these characterization options, consider that the higher activity samples are more likely to identify and quantify the low abundance radionuclides, which are generally required to support the final selection of the ROC (to be discussed later).

2.2.1 Types of Concrete Characterization

2.2.1.1 Traditional Core Bores.

Collecting core bores requires that the boring machine be secured to the floor or wall using concrete anchors. The machine advances a hollow cylinder affixed (3-to-4-inch diameter) with a cutting surface on the concrete side and water is typically used as a cooling agent and lubricant. As the bore is advanced, sections can be broken and withdrawn from the hole and slices can be cut into pucks of the core (typically ½ inch to 1 inch) at desired depths for analysis.

The process of collecting these traditional cores requires substantial labor and is invasive and subject to interferences from rebar and other embedded features. The analysis of the concrete pucks can be done in a few ways including direct analysis of the puck or pulverization of the puck and analysis of the debris. Either method can create analytical errors and anomalies unless carefully controlled.

2.2.1.2 Concrete Dust Collection Through Small Hollow-Core Drilling (TruePro™)

[redacted]

[redacted]

2.2.1.3 Concrete Laser Ablation and Collection

A concrete laser-ablation and collection system was developed in the UK as an innovative tool for material characterization in nuclear applications. It is a sampling tool that can be used in place of conventional drilling to remove material from a surface, to permit the safe, clean, and fast sampling of materials such as concrete, plastic, wood, plaster, and brick for a wide range of radionuclides and for elemental characterization. It can be operated manually or deployed on a remotely operated vehicle.

This technique was developed through a spin-off from an Innovate UK feasibility study in 2013 working in collaboration with Sellafield Ltd. It uses a laser to rapidly (~120 s) collect a small sample (50-100 mg) from a surface and transport the sampled material along a tube (up to 20m long) remote from the active area to a collection point. The laser, control and safety electronics are remote from the sampling head and therefore not affected by local radiation fields. The sample can then be analyzed for any typical suite of radionuclides. The sampling head can be deployed on a remotely operated vehicle and capable of taking samples at heights where scaffolding would normally be required. Sampled material is collected from the surface under vacuum; the sampling process is very clean and does not produce a local dust environment.

During a three-month test of this technology at four sites in the UK, the system collected a total of 350 samples. The collection times per sample were between 2 and 5 min with a throughput rate of less than 10 min. At two of the sites, the system collected material at a height of 8 m on an ROV, work which would have otherwise required scaffolding. Of this population of samples, 100 samples were collected at heights of over 20 m, which demonstrated that the gamma activity for Am-241, Cs137 & Co-60 over the area was below 0.2 Bq/g. The entire process from start to finish took under four hours. Hotspots, previously identified with a beta-sensitive detector, were sampled and analyzed to demonstrate that the activity was due solely to Cs-137 activity, using the gamma activity and total beta counting. In less than two weeks deployment at one of the sites, over 100 samples were taken from walls and floor achieving detection limits of 0.1 Bq/g for alpha activity.

The advantages of this system for characterization are:

- Sampling does not require extensive labor support,
- The sampling method does not produce secondary waste,
- The sampling can be remote from the collection system,
- The sampling method produces uniform surface area sampling profiles, and
- The system can be deployed manually or remotely.

The disadvantages of this sampling method are:

- The sampling mass is small compared to other methods,
- The system requires the use of two support carts, one for the laser generation and the other for the sample collection,
- The system is limited to a sampling depth of approximately 1 mm, and

- The system uses a class 4 laser, although the focusing optics substantially limit the potential laser safety considerations.

2.2.1.4 *In-situ* Gamma Spectroscopy

In-situ gamma ray spectroscopy can be used with solid state (i.e., High Purity Germanium (HPGe)) or scintillation detectors (i.e., NaI(Tl), CeBr, LaBr). Regardless of the detector type, most of these commercially available systems have a detector-specific or generic mathematical efficiency calibration that allows the user to create a source-to-detector model which will then create energy-efficiencies for each of these models. For sites with a limited and defined set of gamma-emitting radionuclides (such as Co-60, Cs-137, Eu-152), a scintillation detector may be a good choice since high-resolution is not needed and higher overall detection efficiencies are typical. For cases where a wide range of radionuclides are present, the solid-state detectors provide better resolution with lower overall detection efficiency while requiring detector cooling at or near liquid nitrogen temperatures to operate.

Regardless of the detector choice, it has been common to place the detectors into shielding collimators such that a defined viewing angle is created, and the user can define the distance to vertical and horizontal surfaces. It has been common for users to define a conservative model of a potential source (small circular area) at the edge of the field of view with the detector positioned 2 to 3 meters from the surface. Using this model, the mathematical efficiencies will convert the identified spectrum peaks to activity per unit area (i.e., pCi/m^2) within the assumed geometry.

As with all other methods, this characterization tool has both advantages and disadvantages.

The advantages include:

- A readily simple and stable deployment and data collection technique,
- Data collection and analysis is relatively quick/inexpensive and does not invoke offsite laboratory analysis costs and delays, and
- Staff operator training can be readily administered.

The disadvantages include:

- The detectors and collimators can be expensive,
- If HPGe is used, a small crane, or equivalent, may be needed for detector positioning,
- Depending on the detector selection, the system can be subjected to gain shifts from large temperature changes,
- The technique is influenced by the ambient gamma radiation levels from discrete and distributed sources,
- The accuracy of the measurement is entirely dependent on the accuracy of the model assumptions, and
- Only gamma-emitting radionuclides are assayed.

2.2.2 Numbers of Concrete Measurements/Samples

[redacted]

[redacted]

[redacted]

2.2.3 Surface Soil Samples

The collection of surface soil samples in impacted areas (0 to 15 cm depth) for characterization purposes should be guided by the information gathered in the HSA so that focused sampling is directed at areas that may have been subject to contamination events. The collection methods should be guided by sampling procedures that can ensure that data quality objectives and measurement quality objectives are met. The number of samples and the area covered are dictated by a subjective analysis of the HSA information and cannot be prescribed.

The analysis of the samples should be performed in a consistent manner and should consider sample preparation (removal of debris, drying and homogenization) and the analysis of HTD radionuclides.

In addition to sampling of impacted areas, this sampling should also consider sampling of regional soils outside of potentially impacted areas to establish a range of activity concentrations resulting from fallout of Cs-137 and/or Sr-90 from the testing of nuclear weapons during the past several decades. This pattern of activity concentrations can be affected by several variables as listed below and the measurements should catalog these.

- Open vs forested areas
- Soil Types
- Soil Origin (Native vs Imported)
- Areal vs Mass-based measurements
- Watershed Characteristics (Drainage ditches, wetlands, leach fields, etc.)

This type of sampling and analysis can provide significant insight into the ranges of background concentrations of radionuclides that can have origins from the site operations or from fallout and should be well understood so that appropriate corrections can be applied.

2.2.4 Subsurface Soil Samples

The collection of subsurface soil samples (depths greater than 15 cm and up to several meters below grade level per MARSSIM) is solely dependent on the potential for current or past subsurface leaks of radioactive material. Therefore, this type of sampling campaign should be focused on likely sources of systems that carried liquid radioactive materials or adjacent to structures where contaminated materials were handled. [redacted]

2.2.5 Radiological Analysis Strategies

2.2.5.1 Selection of Analysis Suites and Establishing the Initial Radionuclides of Concern (ROC)

Selecting an initial suite of radionuclides is important for two purposes. One is to define which radionuclides should be considered for the calculation of DCGLs. The second is to define the ROC list for the offsite laboratory analysis and the associated MDCs that should be included in the laboratory statement of work (SOW) as discussed in MARLAP.

A common approach in defining the initial ROC list is to consider two parallel approaches. The first is to review generic activation data from NUREG/CR-3474 (Reference 4) and NUREG/CR-4289 (Reference 5).

NUREG/CR-3474 provides tables of theoretical activation products for both pressurized water reactors (PWR) and boiling water reactors (BWR) based on typical materials of construction, anticipated impurities, assumed neutron flux, etc. An initial list of radionuclides can be developed from this source showing shutdown radionuclide inventories. However, this initial listing contains many radionuclides typically not present at commercial nuclear power plants.

The second approach is to examine any historical characterization campaigns that may have been used for waste characterization (10 CFR Part 61) or for other operational radiological protection (RP) programs such as EPRI Alpha Monitoring Guidelines (Reference 2) or passive internal monitoring. This data may identify additional radionuclides (or the absence of some) that may otherwise be included in the initial ROC list. This initial list may be used for some of the early DCGL development calculations but should be augmented by the collection of additional samples within specific areas of the plant where elevated radioactive contamination is present.

2.2.5.2 Onsite Sample Analysis

During operations, most sites have an onsite gamma spectroscopy laboratory that is well equipped for the analysis of standard sample geometries such as filters, beakers, or other volumetric samples supported by either NIST-traceable standards or mathematical models (Monte-Carlo based) to determine detection efficiency of a variety of sample geometries that may not be 'standard.' It is common that some of the samples collected during decommissioning characterization can deviate from the standard geometries.

If an onsite laboratory is used for characterization of decommissioning samples, a thorough review of the laboratory capabilities and technical basis should be conducted to ensure that accuracy and precision of these analyses can be demonstrated. This review should include the following:

- Gamma Isotope Library - Review should eliminate short-lived radionuclides and ensure that the library is sufficient to detect radionuclides that may not be abundant during operations but may be observed during D&D due to decay of the short-lived radionuclides. The abundance of short-lived radionuclides is dependent on the time difference between permanent shutdown and characterization measurements and can be site-specific.

- Calibration Standards - Review should ensure that the geometry (mass, dimensions, and sample to detector distance) is appropriate and will not create significant errors when using the calibration standards since some D&D samples will likely differ in these characteristics.

- Mathematical Efficiency Models - If this approach is used for D&D analysis, the site should participate in a periodic analysis comparison program like DOE's MAPEP (Mixed Analyte Performance Evaluation Program) with external sample analysis.

- Detection Sensitivity - Review should ensure that the reported minimum detectable activities (or concentrations), MDAs or MDCs, are well below (no higher than 50% of the DCGL per MARSSIM guidance) the anticipated DCGLs for gamma emitters. This should also include a review of the criteria used to establish a positive detection within the software and how to handle data that is typically reported as "<MDC" and whether other alternatives may be available as discussed in MARLAP.

2.2.5.3 Offsite Sample Analysis

If an offsite commercial laboratory is used for the analysis of characterization samples, a comprehensive statement of work (SOW) should be developed with the laboratory as part of the purchasing vehicle/contract using the MARLAP guidance. The SOW should contain, as a minimum, the following elements.

- QA requirements and laboratory certifications
- Sample matrix types
- MDCs or MDAs for each matrix and analyte
- Sample preservation and quantity requirements
- Analytical reporting requirements for determining detectability
- Reporting results less than the detectability threshold
- Analysis Turn-Around-Time
- QC data reporting requirements associated with each analysis
- Requirements for electronic data deliverables

[redacted]

2.3 Radiological Data Assessment

Whether the analysis of characterization samples is from an onsite or offsite laboratory, the data should be organized to allow for a review of the results from several perspectives as discussed below. However, prior to that level of review, the user should ensure that the data quality objectives have been achieved including:

- The appropriate MDAs or MDCs and uncertainties are sufficient,
- All required analytes have been reported,
- The laboratory QC criteria are met with any exceptions well understood, and
- All sample results are reported.

Once the data has been assessed for quality, the results can be used for determining the appropriate analysis as described in the following subsections.

2.3.1 Identifying Data Trends and Statistical Observations

[redacted]

[redacted]

Equation 2-1

$$A_2(t) = \left(\frac{\lambda_2 A_1(0)}{\lambda_2 - \lambda_1}\right)\left(e^{-\lambda_1 t} - e^{-\lambda_2 t}\right) + A_2(0)e^{-\lambda_2 t}$$

Where: $A_2(t)$ is the activity of Am-241 at decay time t,
$A_1(0)$ is the initial activity (at t=0) of Pu-241
$A_2(0)$ is the initial activity (at t=0) of Am-241
λ_1 is the decay constant for Pu-241
λ_2 is the decay constant for Am-241

2.3.2 Determining Radionuclide Activity Fractions

One of three methods (or any combination or order of these) can be used to determine activity fractions from the analytical data as described below.

1. The first method is to calculate the radionuclide activity fraction, $fA_{i,j,k}$, for each sample, *j*, each radionuclide, *i*, within each population, *k*, from the reported decay corrected radionuclide activity concentrations, $C_{i,j,k}$, using Equation 2-2, and then calculating the average activity fraction, $fA_{i,j,k}$, for each radionuclide, *i*, and population, *k*, of N samples using Equation 2-3.

$$fA_{i,j,k} = \frac{C_{i,j,k}}{\sum(j)C_{i,j,k}}$$ **Equation 2-2**

$$fA_{i,j,k} = \frac{\sum(j)fA_{i,j,k}}{N}$$ **Equation 2-3**

2. The second is to calculate the 75th percentile of the population of samples from Equation 2-2, above. Once the 75th percentile fractions were calculated for each radionuclide, $f_{i,k,.75}$, the data set was re-normalized to determine the percentile-based activity fractions, $fA_{i,k,.75}$ using Equation 2-4.

$$fA_{i,k,.75} = \frac{f_{i,j,k,.75}}{\sum(j)fA_{i,j,k,.75}}$$ **Equation 2-4**

3. The third is to calculate the individual radionuclide ratios to Cs-137 for each sample, $R_{i,Cs\text{-}137,j}$, calculate the 75th percentile for the sample group, $R_{i,Cs\text{-}137,k,.75}$ then renormalize to determine the activity fractions, $f_RA_{i,k,.75}$ using Equation 2-5.

$$f_RA_{i,k,.75} = \frac{R_{i,Cs-137,k,.75}}{\sum(i)R_{i,Cs-137,k,.75}}$$ **Equation 2-5**

Another alternative method to calculate activity fractions is to average the concentrations across a population of sample results for each radionuclide and then calculate the average fraction from this average. However, this method will implicitly weigh the results by activity which may introduce a bias in the activity fractions. Therefore, the analyses described above remove the activity weighting and give equal statistical weight to each of the sample results.

An example of the results of this assessment using all three methods from the analysis of 57 concrete characterization samples for 22 radionuclides is shown in the example below. As noted, there are some differences in the listed radionuclide fractions between the methods and this will be discussed further in the following section.

Table 2-1 Example of Radionuclide Fraction Assessment Results

Nuclide	Average Activity Fractions, fA_i	75 Percentile of the Activity Average Fractions, $fA_{i,,75}$	75th Percentile of the Individual Sample Ratios to Cs-137, $f_RA_{i,75}$
H-3	2.60E-02	2.76E-02	2.52E-02
C-14	9.72E-02	4.43E-02	5.14E-02
Fe-55	7.83E-03	1.42E-03	2.33E-03
Co-58	6.94E-11	4.54E-14	7.00E-14
Ni-59	3.11E-03	1.82E-03	2.48E-03
Ni-63	2.25E-01	2.55E-01	3.43E-01
Co-60	7.79E-03	6.99E-03	1.15E-02
Tc-99	7.09E-03	4.68E-04	5.23E-04
Sr-90	9.22E-03	2.79E-03	2.51E-03
Sb-125	3.12E-04	1.81E-04	1.65E-04
Cs-134	2.05E-04	1.16E-04	1.67E-04
Cs-137	6.08E-01	6.55E-01	5.55E-01
Ce-144	1.91E-05	3.38E-06	3.36E-06
Eu-152	1.35E-03	5.09E-04	7.15E-04
Eu-154	2.33E-03	6.67E-05	1.13E-04
Eu-155	3.45E-04	1.69E-04	2.72E-04
Pu-238	3.86E-04	3.69E-05	3.01E-05
Pu-239/240	9.74E-05	3.01E-05	2.47E-05
Pu-241	4.09E-03	3.19E-03	4.77E-03
Am-241	9.73E-05	1.02E-04	1.32E-04
Cm-243/244	8.54E-05	1.07E-05	1.51E-05
Np-237	7.28E-07	0.00E+00	0.00E+00

2.3.3 Determining Insignificant Radionuclides

NUREG-1757, Vol. 2, Rev. 2 defines radionuclides as “insignificant dose contributors” if the sum of the dose from the group of insignificant contributors (IC) is less than 10% of the total dose from all radionuclides combined. The process for evaluating insignificant contributors needs to be described in the LTP but is not required to be analyzed during the FSS. However, the dose contribution from the insignificant contributors must be accounted for in the final operational DCGLs. The radionuclides remaining after the insignificant contributors are removed are the ROCs for a particular site.

The ROCs should be selected to ensure that sufficient margin has been attributed to the estimated dose contribution from the insignificant radionuclides. This decreases the risk of having to recalculate the adjusted DCGLS.

The Relative Dose Fraction, $RDF_{i,k}$, for nuclide *i* and population *k* is calculated using the site’s DCGLs and the nuclide activity fractions from Equation 2-22.3.2 and Equation 2-6.

$$RDF_{i,k} = \frac{fA_{i,k}}{DCGL_{i,k}}\left[\frac{1}{\sum(i)\frac{fA_{i,k}}{DCGL_{i,k}}}\right] \quad \textbf{Equation 2-6}$$

2.3.4 Final List of Radionuclides of Concern

Some D&D sites have multiple sets of DCGLs to apply for different media or scenarios. This discussion applies to a site showing one radionuclide mix fraction and two sets of DCGLs, one for surface soils (0-15 cm) and one for subsurface soils (down to 1 meter). In this example, the IC dose can be determined using each of the three methods of determining activity fractions and ultimately selecting the most conservative for the final ROC list.

For the example below, in Table 2-2, one mix fraction using the 75th percentile fraction method is used with two sets of DCGLs. In these examples, the Relative Dose Fraction (RDF) is determined for each radionuclide using the applicable DCGLs and summed. This sum represents the RDF. During the iterative selection process, the ROCs were chosen and only those were included in the sum resulting in a relative dose estimate for the ROCs and the IC radionuclides (the remaining nuclides not selected as ROCs). In this example, the ROC dose fraction is 0.992 and 0.976 for surface and subsurface soils, respectively. This leaves the remaining factions (1-f_{ROC}) as the IC dose fraction. As this example analysis shows, only four radionuclides were candidates for ROCs, which account for over 97% of the dose for both surface and subsurface soils.

Table 2-2: Example Determination of the IC Dose Fraction from the 75th Percentile Method Applies to Surface and Subsurface Soils

Nuclide	ROC?	Aux Building Mix Fraction	Soil 0.15 m Relative Dose Fraction	Soil 1.0 m Relative Dose Fraction
H-3		2.52E-02	4.73E-05	3.50E-04
C-14	Y	5.14E-02	1.92E-02	6.05E-02
Fe-55		2.33E-03	1.43E-06	1.32E-06
Co-58		7.00E-14	4.32E-14	2.69E-14
Ni-59		2.48E-03	4.92E-06	1.29E-05
Ni-63		3.43E-01	1.86E-03	4.88E-03
Co-60	Y	1.15E-02	6.49E-02	4.47E-02
Tc-99		5.23E-04	8.64E-05	4.07E-04
Sr-90	Y	2.51E-03	5.06E-03	1.74E-02
Sb-125		1.65E-04	1.39E-04	8.43E-05
Cs-134		1.67E-04	5.83E-04	4.73E-04
Cs-137	Y	5.55E-01	9.06E-01	8.70E-01
Ce-144		3.36E-06	2.75E-07	1.74E-07
Eu-152	Y	7.15E-04	1.81E-03	1.11E-03
Eu-154		1.13E-04	3.07E-04	1.88E-04
Eu-155		2.72E-04	1.98E-05	1.08E-05
Pu-238		3.01E-05	3.86E-06	1.02E-05
Pu-239/240		2.47E-05	3.51E-06	9.31E-06
Pu-241		4.77E-03	1.89E-05	5.50E-05
Am-241		1.32E-04	2.11E-05	5.19E-05
Cm-243/244		1.51E-05	5.01E-06	5.91E-06
Np-237		0.00E+00	0.00E+00	0.00E+00

Nuclide	ROC?	Aux Building Mix Fraction	Soil 0.15 m Relative Dose Fraction	Soil 1.0 m Relative Dose Fraction
		Sum	1.00E+00	1.00E+00
		ROC	9.92E-01	9.76E-01
		IC Dose	8.17E-03	2.39E-02

2.3.5 Surrogate Radionuclides

From the data provided above for an Aux Building, it is evident that Cs-137 is the predominant ROC gamma-emitting radionuclide. Co-60 and Eu-152 are also gamma emitting ROCs that are present, but at much lower fractions than Cs-137. Also, C-14, and Sr-90 are identified as ROCs while also being HTD radionuclides.

Using this data, Cs-137 was selected as the most appropriate gamma emitter for the surrogate relationship for both Sr-90 and C-14. This was based on the high percentage of Cs-137 (55%) and the low percentage of Co-60 (1.15%). Due to the high abundance of C-14 in the mix fractions (~5%), the resulting surrogate ratio to Co-60 would have been impractical to use during additional characterization, remediation, and final status surveys. However, only the Sr-90 surrogate relationship is used here (below) as an example.

The ratio of HTDs to gamma emitters is required to develop a surrogate relationship as defined in MARSSIM. This surrogate relationship allows for the concentration of HTDs to be inferred from the concentration of a gamma emitter in cases where samples are not subject to HTD analysis during FSS activities. Table 2-3 provides decayed concentrations, MDCs and the Sr-90 to Cs-137 ratios for each sample mixture fraction from a characterization data set. (Note: this data set is not intended to be compared with the IC analysis from Table 2-2.) There are some instances where HTDs and/or gamma emitters had reported activity concentrations less than the reported MDCs. In these cases, the MDC was substituted for the concentrations to calculate the Sr-90/Cs-137 ratio.

For the samples where neither nuclide would have been detected, the ratio of the MDCs would not be used since this is merely a ratio of the detectability of the two nuclides for that specific sample and has no relationship to the activity ratio.

Out of the 26 samples, 22 of the Sr-90 results were greater than the MDC, so four samples used the Sr-90 MDC as the reported concentration for the purpose of this example.

Table 12-3: Sr-90 and Cs-137 Decayed Reported Concentrations, MDCs and Ratios

Sample ID	Sr-90 Decayed Reported Concentration (pCi/g)	Sr-90 Decayed MDCs (pCi/g)	Cs-137 Decayed Reported Concentration (pCi/g)	Cs-137 Decayed MDCs (pCi/g)	Sr-90 Concentration Using Reported or MDC Values (pCi/g)	Cs-137 Concentration Using Reported or MDC Values (pCi/g)	Sr-90/Cs-137 Ratio Using Reported or MDC Values
1	3.37E+00	7.15E-01	4.19E+02	4.68E+01	3.37E+00	4.19E+02	8.04E-03
2	7.31E-01	2.47E-01	2.47E+02	2.80E+01	7.31E-01	2.47E+02	2.96E-03
3	1.34E+00	3.77E-01	6.00E+02	4.94E+01	1.34E+00	6.00E+02	2.23E-03
4	3.76E-01	6.63E-02	9.76E+01	2.31E+00	3.76E-01	9.76E+01	3.85E-03
5	1.69E-01	5.33E-02	5.15E+01	1.33E+00	1.69E-01	5.15E+01	3.28E-03
6	2.10E-01	5.75E-02	1.14E+02	2.53E+00	2.10E-01	1.14E+02	1.84E-03
7	4.68E-02	1.94E-01	5.67E+02	4.94E+01	1.94E-01	5.67E+02	3.42E-04

8	1.47E+00	1.21E-01	1.04E+03	5.11E+00	1.47E+00	1.04E+03	1.41E-03
9	1.13E+00	9.77E-02	2.87E+02	3.78E+00	1.13E+00	2.87E+02	3.94E-03
10	6.53E+00	2.46E-01	1.04E+04	1.49E+01	6.53E+00	1.04E+04	6.26E-04
11	7.96E-01	2.93E-01	3.75E+02	3.95E+01	7.96E-01	3.75E+02	2.12E-03
12	1.51E+01	3.37E-01	1.60E+04	1.96E+01	1.51E+01	1.60E+04	9.45E-04
13	1.91E+01	3.92E-01	1.81E+04	2.52E+01	1.91E+01	1.81E+04	1.05E-03
14	1.11E+01	2.08E+00	2.42E+02	2.80E+01	1.11E+01	2.42E+02	4.61E-02
15	1.00E+00	9.09E-02	3.36E+02	2.68E+00	1.00E+00	3.36E+02	2.98E-03
16	1.50E+00	1.10E-01	7.22E+02	4.08E+00	1.50E+00	7.22E+02	2.08E-03
17	2.43E+00	5.73E-01	1.12E+02	1.10E+01	2.43E+00	1.12E+02	2.17E-02
18	4.49E+00	9.29E-01	1.20E+02	1.08E+01	4.49E+00	1.20E+02	3.74E-02
19	-3.37E-02	4.17E-02	3.27E+01	1.12E+00	4.17E-02	3.27E+01	1.28E-03
20	1.54E+01	3.54E-01	6.48E+03	1.21E+01	1.54E+01	6.48E+03	2.37E-03
21	2.01E+01	4.12E-01	8.99E+03	1.40E+01	2.01E+01	8.99E+03	2.24E-03
22	1.04E+01	1.98E+00	5.10E+03	6.03E+02	1.04E+01	5.10E+03	2.05E-03
23	5.50E+00	2.17E-01	3.09E+03	7.85E+00	5.50E+00	3.09E+03	1.78E-03
24	3.15E+00	3.20E-01	6.68E+02	5.04E+00	3.15E+00	6.68E+02	4.71E-03
25	1.67E-01	2.06E-01	4.60E+01	6.10E+00	2.06E-01	4.60E+01	4.47E-03
26	4.65E-02	4.66E-02	5.80E+01	1.53E+00	4.66E-02	5.80E+01	8.03E-04

Table 2-4: Sr-90/Cs-137 and Associated Statistical Parameters Based on Reported or MDC Concentrations

Parameter	AB/TB/RWPB Sample Population
Average	2.95E-02
Minimum	3.42E-04
Maximum	3.87E-01
Standard Deviation	8.96E-02
% Coefficient of Variation	303.37%
75th Percentile	4.71E-03
95th Percentile	1.97E-01

In Table 2-4, the surrogate ratio of Sr-90 to Cs-137 was chosen as the 95^{th} percentile, or a value of 0.197 and in this case, a similar approach can be selected for C-14.

2.4 Other Use of Site Characterization Data

[redacted]

Plans for the performance of the site characterization are developed based on the results of a Historical Site Assessment (HSA) as described in the MARSSIM guidance. The HSA helps to determine the extent and nature of the contamination at the site by reviewing incidents that occurred during the operation of a plant. As required by 10 CFR Part 50.75 (g), a listing of plant occurrences such as spills and other contamination incidents needs to be maintained during the operation of the plant and during decommissioning.

It is recommended that a thorough site characterization be performed early in decommissioning if not prior to permanent shutdown. The results of a site characterization may affect how the decommissioning is conducted. Completion of the initial site characterization allows the site:

- To divide the site into manageable sections or areas for survey and classification purposes.

- To identify the potential and known sources of radioactive contamination in systems, on structures, in surface or subsurface soils, and in ground water.
- To determine the initial MARSSIM classification of each survey area.
- To develop the initial radiological and hazardous material information to support decommissioning planning including building decontamination, demolition, and waste disposal.
- To develop the information to support Final Status Survey design including instrument performance standards and quality requirements, radionuclide ratios.
- To identify any unique radiological or hazardous material health and safety issues associated with decommissioning.

Initial site characterization efforts should include unaffected structures and land areas to ensure that all locations are surveyed to verify that plant-related contamination is not present in these areas. As significant survey data is likely available in known affected areas, less characterization is needed in these areas.

2.4.1 Initial Survey Area Classification

After sufficient characterization information has been collected, the initial survey area classification can be performed. It may be efficient to establish survey area boundaries using logical physical boundaries and site landmarks. Many areas will be further subdivided into survey units. A survey unit is a physical area consisting of structures or land areas of specified size and shape for which a separate decision is made as to whether residual contamination in that area exceeds the release criterion. Survey areas for subsurface soils include any sub-surface features such as piping and drain systems.

Classification of a survey area has a minimum of two stages: (1) initial classification and (2) final classification. Initial classification is performed only once, at the time of identification of the survey unit using the information available. Final classification is performed and verified as an objective of the Final Status Survey plan. When additional information is obtained during the decommissioning process through Characterization Surveys or Remedial Action Surveys (performed to track the effectiveness of decontamination techniques), the data can be assessed using the Data Quality Objective (DQO) process to verify that the current classification of the survey area is appropriate, to guide reclassification of the survey area, and/or to guide the design of subsequent surveys. Approved site procedures should be prepared to govern the process of classification and mandate appropriate documentation of the classification results.

It is recommended that a realistic assessment of the available data be used in setting the initial survey area classifications. As these classifications are included in the LTP for a site, typically the NRC requires staff approval of any change of a survey area to a lower classification. As this may delay the conduct of an FSS while a justification is prepared and the NRC review is completed, many sites have decided not to downgrade the classification of any survey areas. As increases to classifications do not require NRC approval, it is recommended that initial classifications be realistic with classifications being adjusted upward (i.e., Class 3 to Class 2 or Class 2 to Class 1) prior to and during FSS activities as warranted by additional sample data.

The definitions of Class 1, 2, and 3 (per MARSSIM) are as follows:

- Class 1 Areas: Areas that have, or had prior to remediation, a potential for radioactive contamination (based on site operating history) or known contamination (based on previous radiation surveys) above the $DCGL_W$. Examples of Class 1 areas include: 1) site areas previously subjected to remedial actions, 2) locations where leaks or spills are known to have occurred, 3) former burial or disposal sites, 4) waste storage sites, and 5) areas with contaminants in discrete solid pieces of material and high specific activity.

- Class 2 Areas: Areas that have, or had prior to remediation, a potential for radioactive contamination or known contamination, but are not expected to exceed the $DCGL_W$. Examples of areas that might be classified as Class 2 for the Final Status Survey include: 1) locations where radioactive materials were present in an unsealed form, 2) potentially contaminated transport routes, 3) areas downwind from stack release points, 4) upper walls and ceilings of buildings or rooms subjected to airborne radioactivity, 5) areas handling low concentrations of radioactive materials, and 6) areas on the perimeter of former contamination control areas.

- Class 3 Areas: Any impacted areas that are not expected to contain any residual radioactivity or are expected to contain levels of residual radioactivity at a small fraction of the $DCGL_W$, based on site operating history and previous radiation surveys. Examples of areas that might be classified as Class 3 include buffer zones around Class 1 or Class 2 areas, and areas with very low potential for residual contamination, but insufficient information to justify a non-impacted classification.

- Non-Impacted: Areas that have no potential for residual radioactivity. An example of non-impacted areas is one that is well away from areas where residual radioactivity has been found and where no activities that have the potential for spreading contamination have occurred.

Class 1 areas have the greatest potential for contamination and therefore receive the highest degree of survey effort for the Final Status Survey, followed by Class 2 areas, and then by Class 3 areas. Non-impacted areas do not require any level of survey coverage because they have no potential for residual contamination. The Final Status Survey plans should include a process by which measurements that approach investigation levels, defined as fractions of the DCGLs, are reviewed to see if reclassification of an area(s) is necessary.

2.4.2 Understanding the Extent of Contamination

The extent of contamination in structures and land areas on a site must be known in order to provide input to the remediation planning process. This information is collected through site characterization. It should be noted that the site characterizing process is iterative with additional surveys being conducted as decommissioning progresses.

If characterization surveys are used to collect the information needed to support remediation assessments, a carefully designed survey plan is needed. One of the key parts of designing the measurements and sampling of an effective survey plan is to establish and document Data Quality Objectives (DQOs), or an equivalent strategy to be used to directly determine the extent of contamination. The surveys may include sampling to determine the horizontal and vertical extent of the contamination.

3 IDENTIFICATION OF REMAINING SITE DISMANTLEMENT ACTIVITIES

3.1 Introduction

Chapter 3 describes the remaining site dismantlement activities. The regulation applicable to this area of review is 10 CFR 50.82(a)(9)(ii)(B).

A significantly detailed schedule of remaining dismantlement and remediation activities should be included in this section. Planned activities to be addressed include, but are not limited to, building demolitions, remediation of standing structures to remain after license termination, soil/ground water remediation, and site restoration.

Chapter 3 includes a discussion of important regulatory issues such as requirements for coordination with other federal and/or state regulatory agencies and the process for doing so. Per RG 1.179 the LTP should list the remaining activities that do not involve unreviewed safety questions or changes in a facility's technical specifications, with sufficient detail for the staff to confirm that remedial activities may be performed under 10 CFR 50.59.

This chapter also includes important details that should be obtained from site engineering and D&D regarding the radiological impacts of decommissioning activities; to ensure appropriate administrative and engineering controls are in place throughout the D&D process to prevent the contamination of previously unimpacted areas, and the recontamination of remediated or surveyed areas. The key messaging is assurance that radioactive contamination will be controlled and maintained at the source. There are many lessons learned across the industry where contamination events occurred resulting in either localized or widespread contamination across open land areas of a site, including release of discrete radioactive particles (DRPs). High risk activities include waste loadout, physical or torch cutting of contaminated components without proper negative ventilation, heavy equipment/vehicle movement, and environmental processes (rain, wind, drainage, etc.).

Chapter 3 also requires a discussion on occupational and public exposure to contamination/radioactivity. Radiological Operations/Protection personnel should provide estimated exposure to workers broken down by major activities (e.g., reactor vessel head work, asbestos removal, waste operations, routine walkdowns, ISFSI, etc.).

Projected waste quantities showing the type of waste, the waste classification, weight, packing density and waste volume are also included in this chapter.

Note that RG 1.179 states "the details in this section should be sufficient for the NRC to identify any inspection or technical resources needed during the remaining dismantlement activities."

All of the above underscore the importance of engaging those site personnel with the experience and expertise (D&D, engineering, waste, industrial hygiene, regulatory affairs, etc.) to develop required details of chapter 3.

In the following sections, the focus should be the effect these activities may have on the necessary radiological control procedures (differences from operations vs. decommissioning activities).

3.2 Radiological Control Procedures

Based on activities planned for reactor vessel work, systems removal, structures removal, remediation techniques and site restoration, indicate any necessary additions or modifications to the radiological control procedures to support the decommissioning and license termination of the site. Radiological procedures to support final site surveys may be included in chapter 5 and referenced within this chapter.

3.3 Structures at License Termination

Provide a table of site structures denoting the plans for either removing or the remaining of site structures after license termination, including the time frame for structures to be removed. For structures to remain, indicate requirements for remediation and time frame for such remediation.

3.4 Soil and Groundwater Remediation

Indicate potential or known needs for remediation of soils and/or groundwater. If none is anticipated, state the basis for the assumptions. For known remediation activities that will occur in these areas, indicate if current radiological control procedures are adequate or addressed in section 3.2 above.

3.5 Waste Disposal Plans

Information in this section is from EPRI Report, "Basis for National and International Low Activity and Very Low-Level Waste Disposal Classifications" (Reference 7).

3.5.1 Disposal at NRC Licensed Facilities

There are currently disposal options available to U.S. plants for all Class A, B and C wastes at commercially operated surface or near-surface disposal facilities. Operation of most of these facilities is licensed by the NRC. However, there are some state licensed hazardous/radioactive waste landfills that do not require an NRC license to operate. These landfills can receive waste with radioactivity levels in an approximate range of 1 to 40 pCi/g (0.04 to 1.5 Bq/g) of gamma radionuclides.

3.5.2 Disposal at Hazardous Waste Landfill Licensed to Receive NRC Exempted Radwaste

Much of the building demolition waste and soil remediation waste from decommissioning contains very low levels of radioactivity. The NRC has a process where applications can be made for the exemption of waste with very low levels of radioactivity from the requirement for disposal at an NRC licensed facility. The application to the NRC must show that this disposal will not result in any member of the public receiving a dose of more than 5 mrem/yr (0.05 mSv/yr). The Humboldt Bay Power Plant has recently received approvals for disposal of decommissioning waste at such a facility through this process.

3.5.3 NRC Waste Exemption Process

The NRC uses a case-by-case exemption process to provide an alternative disposal option for radioactive materials. This process was rarely used by the nuclear power industry prior to the 1990s. Additionally, waste processors in Tennessee have been using a process like the 10 CFR 20.2002 exemption process for processing increasing volumes of operating and decommissioning nuclear power plant wastes. This approach has evolved into Tennessee's Bulk Survey for Release (BSFR) Program.

Prior to 2000, most 10 CFR 20.2002 requests were for disposal of the waste onsite. Between 2000 and 2006, the NRC received 20 requests for alternative disposal. 85% of those requests were for offsite disposal. Increasingly, RCRA hazardous waste facilities were the off-site, alternative disposal location being requested. In those instances, state regulators responsible for the operation of the RCRA disposal facility were involved. Additional information on this topic is contained in EPRI Report, “Generic Technical Basis for Implementing a Very Low-Level Waste Category for Disposal of Low Activity Radioactive Wastes” (Reference 8).

3.5.4 Other Radioactive Waste Considerations

[redacted]

[redacted]

3.6 Schedule

Provide a detailed schedule for remaining decommissioning, site restoration, Final Status Survey, and request for license termination activities. This is typically a “snapshot” of planning software tools along with acknowledgement that as schedules may change, periodic updates will be provided to the NRC.

4 REMEDIATION PLANS

4.1 Introduction and Background

This chapter of the LTP discusses in detail how facility and site areas will be remediated to meet the NRC's release criteria. The regulations applicable to this area of review are 10 CFR 50.82(a)(9)(ii)(C) and Subpart E of 10 CFR Part 20.

Chapter 4 includes a discussion of the remediation methods and techniques that the licensee will use to demonstrate that the facility and site areas meet the NRC criteria for license termination in Subpart E of 10 CFR Part 20. As discussed in the introduction for chapter 3, chapter 4 also requires input from site experts performing D&D and implementing appropriate radiological controls

The remediation methods that will be employed for various media (e.g., scabbling, hydrolazing, needle guns, grit blasting, etc.) should be described in detail, as well as how these activities will impact radiological conditions.

Chapter 4 also includes an ALARA evaluation using guidance provided in Appendix N of NUREG-1757, Volume 2, which describes acceptable methods for determining when further reduction of residual activity is required to concentrations below the levels necessary to satisfy the 25 mrem/year dose criterion.

Note that the NRC does not require the LTP to include details regarding changes to the radiation protection program, but periodic updates to the final safety analysis report should provide such details.

Information in this chapter is taken from the EPRI FSS Experience Report (Reference 6) unless otherwise noted.

4.2 Lessons Learned

[redacted]

- [redacted]
- [redacted]
- [redacted]

4.3 Remediation Levels and ALARA Evaluations

When dismantlement and decontamination actions are completed, residual radioactivity may remain on building surfaces and on-site soils. Residual radioactivity must satisfy the provisions of 10 CFR 20, Subpart E. As depicted in Figure 4-1, the ALARA cleanup levels for a power plant decommissioning may be established at one of two levels:

- A predefined generic ALARA screening, or
- A survey unit specific ALARA evaluation.

In either case, the ALARA evaluation uses an action level, referred to as a remediation level. This remediation level corresponds to a residual radioactivity concentration at which the averted collective radiation dose converted into dollars is equal to the costs of remediation (e.g., risk of transportation accidents converted into dollars, worker and public doses associated with the remediation action converted into dollars, and the actual costs to perform the remediation activity).

If the value of further dose reduction from remediation is greater than the “costs” of the action, then the remediation action being evaluated is cost-effective and should be performed. Conversely, if the value of further dose reduction is less than the costs, the levels of residual radioactivity are considered ALARA and therefore further remediation action would not be required. The methodology and equations used for performing ALARA evaluations should be consistent with Appendix N of NUREG-1757, Volume 2, “Consolidated Decommissioning Guidance” (Reference 9). Documentation of ALARA evaluations must be included in the Final Status Survey report for each survey area.

4.3.1 Generic ALARA Screening Levels

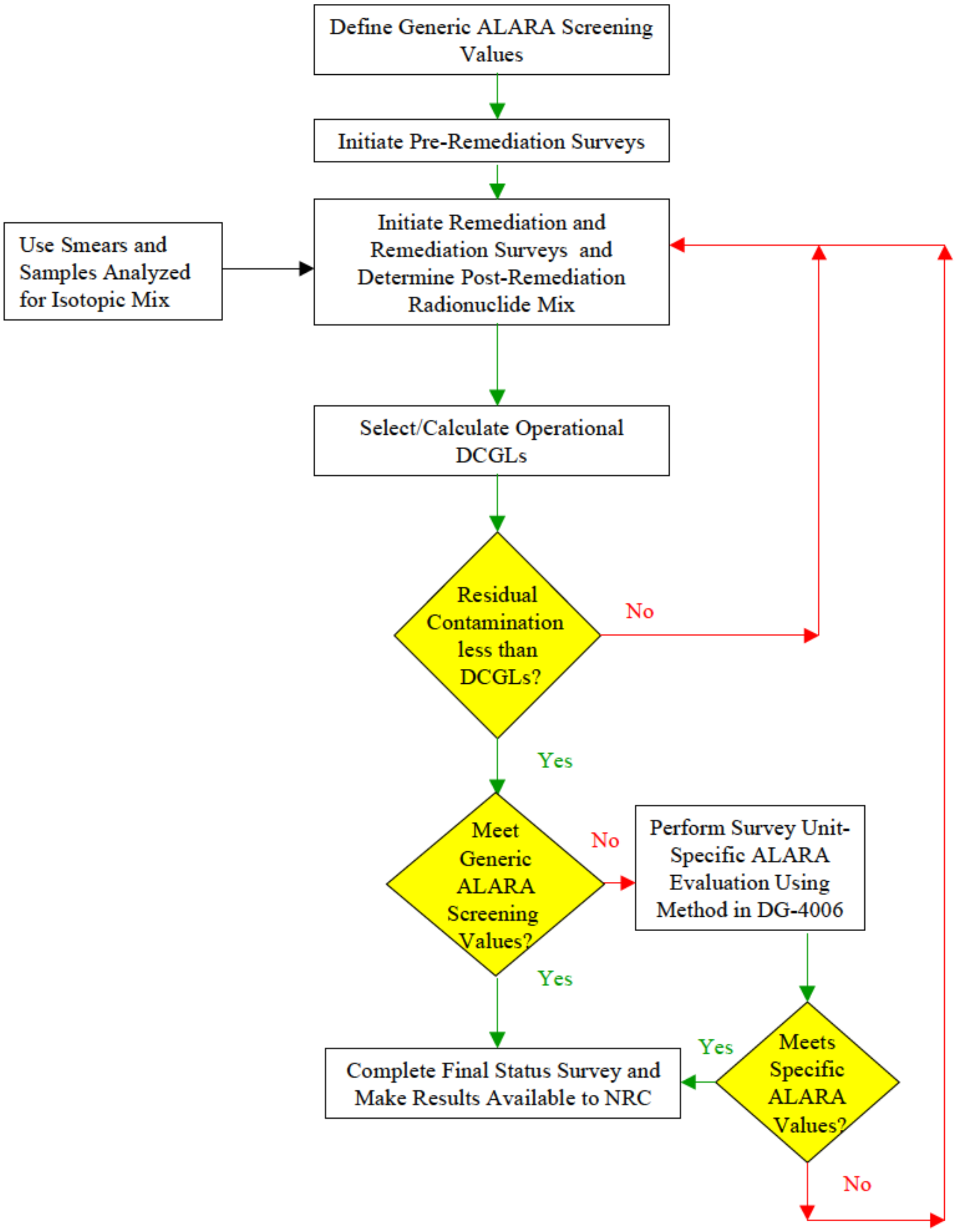
As discussed in Appendix N of NUREG-1757, Volume 2, soil remediation beyond the DCGLs is not likely to be cost-beneficial due to the high costs of waste disposal. This has been confirmed by ALARA evaluations performed at several decommissioning sites.

For building surfaces, a generic ALARA screening value could be calculated using conservative estimates for building remediation costs. Appendix N of NUREG-1757, Volume 2, also contains guidance on performing ALARA evaluations for building structures.

For building surfaces, a generic ALARA screening value could be calculated using conservative estimates for building remediation costs. This generic ALARA screening value can be calculated using the guidance in Appendix N of NUREG-1757, Volume 2, after additional characterization has been undertaken and remediation methods have been evaluated for their effectiveness. This value can represent the level expressed as a percentage or fraction of the DCGL (NRC generic screening or site specific as discussed in Chapter 6), for which the benefit of further remediation of structures is greater than the associated costs.

Upon completion of post-remediation surveys and satisfaction of the 25 mrem/yr (0.25 mSv/yr) TEDE criteria, the level of residual radioactivity in the survey area can be compared against the appropriate generic ALARA screening value. Where the level of residual radioactivity is lower than the generic ALARA screening value, the remediation is clearly ALARA, no further remediation is required, and final status surveys can proceed. Where the level residual radioactivity is greater than the generic ALARA screening value, a survey-unit ALARA evaluation is performed to determine the survey unit-specific ALARA remediation level for comparison.

Figure 4-1
Survey Unit ALARA Evaluation Process



[redacted]

[redacted]

4.3.2 Groundwater ALARA Evaluation

As discussed in Appendix N to NUREG-1757, Volume 2, if there is residual radioactivity from site operations in groundwater, it may be necessary to calculate the collective dose from consumption of the groundwater. Dose modeling, such as that discussed in chapter 4 for Connecticut Yankee, assumes that the aquifer does not supply a large population, but only the resident farming family. However, if it is determined that drinking water for a large population could be supplied by groundwater onsite, the collective dose for that population would need to be included in the ALARA calculation as indicated in NUREG-1757, Volume 2, Appendix N (Reference 9).

4.4 Techniques & Approaches to Remediating Structures, Soils, and Groundwater

Remediation actions may be required to reduce the residual radioactivity levels below the applicable cleanup criteria as provided in section 4.3. The specific remedial actions depend on the type of area under consideration. These area types are categorized as one of the following:

- Structures (including building interiors and exteriors, major freestanding exterior structures, exterior surfaces of plant systems, and paved exterior ground surfaces);
- Soils; and
- Nonstructural plant systems (including interior surfaces of process piping and components).

4.4.1 Structures

[redacted]

[redacted]

4.4.2 Shallow Remediation Techniques

[redacted]

- [redacted]
- [redacted]
- [redacted]

4.4.3 Aggressive Remediation Techniques

[redacted]

[redacted]

- [redacted]
- [redacted]
- [redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

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[redacted]

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[redacted]

[redacted]

4.4.4 Soils

[redacted]

[redacted]

[redacted]

[redacted]

4.4.5 Soil Mixing

[redacted]

Appropriateness of Allowing Intentional Mixing - Although some mixing of uncontaminated and contaminated soils inevitably occurs during the course of a remediation, the NRC has not permitted intentional mixing of contaminated soil with non-contaminated soil. The Commission approved the staff recommended options allowing the evaluation for approval of the following:

- The resultant footprint of the area containing the contaminated soil after license termination should be equal to or smaller than the footprint of the zones of contamination before the decommissioning work begins.

- Clean soil, from outside the footprint of the area contaminated soil, should not be mixed with contaminated soil to lower concentrations although the use of soil from outside the footprint will be considered in rare cases.

- The RIS states that cases will only be considered where it can be demonstrated that removal of the soil would not be reasonably achievable.

[redacted]

4.4.6 Nonstructural Systems

Contaminated plant systems and components may be sent to an offsite processing facility or to a low-level radioactive waste disposal facility. Slightly contaminated systems may be decontaminated onsite and released. Nonstructural systems and components generally are surveyed and released using existing plant procedures and processes (i.e., “free release criteria”).

4.5 Ongoing Contamination Control of Remediated Areas & Equipment

Isolation and control measures need to be implemented through approved plant procedures and need to remain in force throughout final survey activities until there is no risk of contamination from decommissioning or the survey area has been released from the license. If isolation and control measures established for a given survey unit are compromised, evaluations need to be performed and documented to confirm that no radioactive material was introduced into the area that would affect the results of the Final Status Survey.

To provide additional assurance that land areas and any structures that have successfully undergone FSS remain unchanged until final site release, documented periodic evaluations of the FSS areas should be performed. The periodic evaluation can consist of:

- A walkdown of the areas to check for proper postings,

- Check for materials introduced into the area since the last evaluation,

- Any general disturbance that could change the FSS including the potential for contamination from adjacent decommissioning activities,

- A review of the recent incident files maintained by the site (may be contained in files maintained per NRC regulation 10CFR50.75(g)(1)).

- For structures, the use of routine Health Physics surveys.

5 FINAL RADIATION SURVEY PLAN

This section will discuss the standard techniques used to conduct Final Status Surveys at nuclear plants being decommissioned. Information in this Chapter is taken from the EPRI FSS Experience Report (Reference 6).

It should be noted that this chapter contains values for certain input parameters that were used in the dose modeling computer codes by certain licensees. The reader needs to be aware that these types of values may be site-specific and may not be appropriate for their site.

The regulations applicable to this area of review in 10 CFR 20.1402 are found in 10 CFR 50.82(a)(9)(ii)(D) and 10 CFR 20.1501(a) and (b).

The two radiological criteria for unrestricted use specified in 10 CFR 20.1402 are: 1) the residual radioactivity that is distinguishable from background radiation results in a total effective dose equivalent (TEDE) to an average member of the critical group (AMCG) that does not exceed 25 millirem/year (mrem/year), including that from groundwater sources of drinking water, and 2) the residual radioactivity has been reduced to levels that are as low as is reasonably achievable (ALARA).

The MARSSIM (NUREG-1575) and the NRC Consolidated Guidance (NUREG-1757) describe Final Status Surveys (FSS) as radiological surveys performed to demonstrate compliance with dose or risk-based regulatory requirements for the restricted or unrestricted release of a property following site cleanup (D&D and remediation). In almost all cases a site is released for unrestricted use via a Partial Site Release (PSR), given that the stored fuel in the ISFSI remains on the site (this part of the site must remain under an NRC Part 50 license).

The FSS process is technically challenging. FSS guidance documents are complex and lengthy and require specific expertise for successful implementation. There are also accompanying specialized software and statistical methods that are critical to designing the FSS.

The FSS Plan is required as chapter 5 of the site's License Termination Plan (LTP) and therefore is a license condition that must be followed.

The objective of this document is to provide a high-level survey summary of the FSS process, emphasizing the importance of compliance.

The primary objectives of the FSS are to:

- Verify the survey unit classification.
- Demonstrate that the potential dose from residual radioactivity in each survey unit is below the release criterion.
- Demonstrate that the potential dose from small areas of elevated activity, when combined with other residual radioactivity in a SURVEY UNIT, is below the release criterion.

There are some high-level terms that are important to understand for FSS. Those include survey unit, survey unit classification, DCGLs, survey package, and sample plan.

A survey unit is a geographical area consisting of structures, land areas, or buried piping of specified size and shape for which a separate decision will be made of whether the survey unit meets the radiological release criteria. Survey units are contiguous site areas, with a similar use history and the same classification of contamination potential. Survey units are established to facilitate the FSS process and the statistical analysis of the data.

Survey unit classifications are Class 1, Class 2, or Class 3. Class 1 survey units have the highest potential for contamination approaching the release criteria (DCGLs) and Class 3 has the lowest potential. Survey units or areas of the site can also be classified as "non-impacted" meaning they do not require FSS but do require a justification for the classification during the site characterization process.

Survey units have a size limit, with Class 1 survey units being the smallest and Class 3 being the largest. The survey size requirements for each classification are provided in Table below.

Table 5--1
Survey Unit Surface Area Limits

Survey Unit Classification	Surface Area Limit
Class 1: Structures (floor area) Land areas	 100 m^2 2,000 m^2
Class 2: Structures (floor area) Land areas	 1,000 m^2 10,000 m^2
Class 3: Structures (floor area) Land areas	 no limit no limit

DCGLs are the site-specific release criteria for each media type. DCGLs are radionuclide-specific and are equivalent to the level of residual radioactivity (above background levels) that could result in a total effective dose equivalent (TEDE) of 25 mrem per year to a member of the public (or average member of the critical group). A detailed discussion of the different types of DCGLs is provided in section 5.1.2. However, a typical commercial reactor site has multiple source terms (types of contaminated media) – including surface soil, subsurface soil, surface and subsurface structures, buried pipe, etc. Therefore, the base case DCGLs are ratioed (reduced) for each media type to ensure that the summation of dose from all source terms is less than 25 mrem/year (or applicable site-specific LTP compliance dose) after all FSS is complete.

DCGLs equate to the concentration levels that represent the maximum annual dose criterion of 10 CFR 20.1402 (typically 25 mrem per year to a member of the public). The process for developing DCGLs is very complex and is typically performed using a software package called RESRAD. The RESRAD family of codes was developed and is maintained by Argonne National Laboratory. Refer to chapter 6 for further explanation of RESRAD and DCGLs.

The FSS process involves:

- survey design
- survey implementation
- data assessment

5.1 Standard Final Site Survey (FSS) Techniques

The first step in conducting an adequate FSS is to design the survey. Per NRC guidance contained in the Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM, Reference 11), the objectives and steps of that design need to be developed using the Data Quality Objectives (DQO) process. The following illustrates the DQO process being used at several sites to design an FSS.

5.1.1 Data Quality Objectives

The DQO process is incorporated as an integral component of the data life cycle for a plant being decommissioned. The DQO process is used in the planning phase for scoping, characterization, remediation, and FSS plan development using a graded approach. Survey plans that are complex or that have a higher level of risk associated with an incorrect decision (such as FSS) require significantly more effort than a survey plan used to obtain data relative to the extent and variability of a contaminant. This process, described in MARSSIM, is a series of planning steps found to be effective in establishing criteria for data quality and developing survey plans. DQOs allow for systematic planning and are specifically designed to address problems that require a decision to be made and provide alternate actions. Furthermore, the DQO process is flexible in that the level of effort associated with planning a survey is based on the complexity of the survey and nature of the hazards. Finally, the DQO process is iterative, allowing the survey planning team to incorporate new knowledge and modify the output of previous steps to act as input to subsequent steps.

The DQO process consists of performing the following seven steps:

- State the Problem
- Identify the Decision
- Identify the Inputs to the Decision
- Define the Boundaries of the Decision
- Develop a Decision Rule
- Specify Tolerable Limits on Decision Errors
- Optimize the Design for Obtaining Data

The actions that can be taken to address these DQO process steps during the planning of an FSS for a particular survey area are addressed below:

- State the Problem

The first step of the planning process consists of defining the problem. This step provides a clear description of the problem, identification of planning team members (especially the decision-makers), a conceptual model of the hazard to be investigated, and the estimated resources. The problem associated with FSS is to determine whether an area meets the NRC's radiological criterion for unrestricted release (Reference 12).

- Identify the Decision

This step of the DQO process consists of developing a decision statement based on a principal study question (i.e., the stated problem) and determining alternative actions that may be taken based on the answer to the principal study question. Alternative actions identify those measures to resolve the problem. The decision statement combines the principal study question and alternative actions into an expression of choice among multiple actions. For FSS, the principal study question is normally "Does residual radioactive contamination present in the survey unit exceed the release criteria?" The alternative actions may include no action, investigation, resurvey, remediation, and reclassification.

- Identify Inputs to the Decision

The information required here depends on the type of media under consideration (e.g., soil, water, concrete) and whether existing data are sufficient or new data are needed to make the decision. If the decision can be based on existing data, then the source(s) will be documented and evaluated to ensure reasonable confidence that the data are acceptable. If new data are needed, then the type of measurement (e.g., scan, direct measurement, and sampling) is determined. Sampling methods, sample quantity, sample matrix, type(s) of analyses and analytic measurement process performance criteria, including detection limits, are established to ensure adequate sensitivity relative to the action level and to minimize bias. Action levels provide the criterion for choosing among alternative actions (e.g., whether to take no action, perform confirmatory sampling).

- Define the Boundaries of the Study

This step of the DQO process includes identification of the target population of interest, the spatial and temporal features of the population pertinent to the decision, time frame for collecting the data, practical constraints and the scale of decision making. In FSS, the target population is the set of samples or direct measurements that constitute an area of interest (i.e., the survey unit). The medium of interest (e.g., soil, water, concrete, steel) is specified during the planning process. The spatial boundaries include the entire area of interest including soil depth, area dimensions, contained water bodies and natural boundaries, as needed.

- Develop a Decision Rule

This step of the DQO process develops the binary statement that defines a logical process for choosing among alternative actions. The decision rule is a clear statement using the "if...then..." format and includes action level conditions and the statistical parameter of interest (e.g., mean of data). Decision statements can become complex depending on the objectives of the survey and the radiological character of the affected area.

- Specify Tolerable Limits on Decision Errors

This step of the DQO process incorporates hypothesis testing and probabilistic sampling distributions to control decision errors during data analysis. Hypothesis testing is a process based on the scientific method that compares a baseline condition to an alternate condition. The baseline condition is technically known as the null hypothesis. Hypothesis testing rests on the premise that the null hypothesis is true, and that sufficient evidence must be provided for rejection.

The primary consideration during FSS is demonstrating compliance with the release criteria. The following statement may be used as the null hypothesis for an FSS: “The survey unit exceeds the release criteria.”

5.1.2 Radiological Release Limit Terminology

Prior to discussing FSS planning and implementation, some definitions of site release limit terminology are needed:

DCGL – Derived Concentration Guideline Level: Per MARSSIM (Reference 11) these are the radionuclide concentrations in media or building surface activity levels against which actual measurements will be compared to determine if the Survey Area/Unit (the land area or room in a building over which the Final Status Survey is being conducted) meets the dose limits of the LTR or other lower limits established by the stakeholders. There are a few specific types of DCGLs as described in the following:

$DCGL_W$ - The $DCGL_W$ is the level that the **average** residual radioactivity soil concentration or surface contamination level in a complete survey unit cannot exceed for the survey area/unit to be released for unrestricted use at license termination.

$DCGL_{EMC}$ - $DCGL_{EMC}$ values are those obtained when the $DCGL_W$ values are scaled to obtain a level that represents the same dose to an individual from residual contamination over a smaller area within a survey unit. The subscript EMC stands for elevated measurement comparison. The $DCGL_{EMC}$ is defined as the product of the applicable $DCGL_{W,}$ and a correction factor known as the area factor. The area factor is equal to the ratio of the dose calculated when the total area of the survey unit is assumed to be at the $DCGL_{EMC}$ contamination level to the dose when a smaller contaminated area is assumed to have the $DCGL_{EMC}$ contamination level.

Area factors are required to be determined for use in final status surveys. Area factors for activity in soil can be determined using the D&D or RESRAD codes. Area factors for surface activity on buildings can be determined using the D&D or RESRAD-Build codes. These codes are described in chapter 6.

5.1.3 Other Aspects of FSS Planning

As decommissioning proceeds, areas will, as necessary, be decontaminated to remove loose surface contamination (as well as fixed contamination) from building surfaces or, in the case of land areas, soil will be removed to levels that meet the appropriate DCGLs. When the radionuclide concentrations in an area have been shown by characterization data or, in the case of an area that has been remediated, by the Remedial Action Survey (per MARSSIM, Reference 11) to be below the required DCGLs, it is released for FSS. This release involves the turnover of the physical and administrative control of the area to FSS personnel for preparation, design, and performance of the FSS.

For each survey type (characterization, remedial action, and final status) a documented survey plan should be developed using the DQO process. The level of effort with which the DQO process is used as a planning tool is commensurate with the type of survey and the necessity of avoiding a decision error. This is the graded approach of defining data quality requirements. For example, scoping and characterization survey plans intended to collect data might only require a survey objective and the instrumentation and analyses specifications necessary to meet that survey objective. Remedial action and final status survey plans, which typically require decisions, need additional effort during the planning phase according to the level of risk of making an error and the potential consequences of that error. These survey plans should contain the appropriate data assessment to ensure that several objectives are met. These objectives include:

- Appropriate instrument selection to ensure the proper sensitivity relative to the applicable DCGLs,
- Appropriate instrument quality control measures to ensure operability,
- Appropriate survey techniques, as described in NUREG-1507 (Reference 13), to ensure that the field measurement techniques are consistent with the calibration methodologies,
- Appropriate sample collection and analysis to determine spatial variability and variability in radionuclide ratios,
- Data analysis criteria to identify follow-up actions such as remediation and the collection of additional samples, and
- Appropriate classification of survey area.

5.2 Building Surveys

The survey methods to be employed in FSS can consist of combinations of advanced technologies, scanning, fixed measurements, sampling, and other methods as needed to meet the survey objectives.

5.2.1 Scanning

Scanning is a widely used process to investigate contamination levels on building surfaces. During scanning, the operator uses portable radiation detection instruments to detect the presence of radionuclides on a specific wall or floor surface. The term scanning survey is used to describe the process of moving portable radiation detectors across a surface with the intent of locating residual radioactivity. This process is often conducted by trained technicians but can be performed by equipment designed to move the same type of portable detector in a controlled manner while recording readings during the process of the detector travel.

Investigation levels for scanning surveys are determined during survey planning to identify areas of elevated activity. Scanning surveys are performed to locate radiation anomalies indicating residual gross activity that may require further investigation or action.

Numerical limits for release of building and other structures, the DCGLs, are in terms of surface contamination levels. Units are generally in disintegrations per 100 cm^2. Beta radionuclide scanning instrumentation is generally used to conduct these types of surveys. More recently, gamma sensitive equipment has been used to detect contamination in cracks and crevices. Measurements need to be taken both during scanning and during fixed measurement with the instrument stationary, although some advanced technologies can achieve these two surveys in one step.

5.2.1.1 Instrument Sensitivity

The selection of appropriate instrumentation for post-remediation surveys is important from a planning and financial risk management perspective. For post-remediation surveys that are conducted to verify that the remediation target is achieved, FSS instrumentation does not need to be used. Although, the use of the actual FSS instrumentation to evaluate remedial actions is preferred as this will help to ensure that the remediated area passes the subsequent FSS.

Instrument detection limits are typically quantified in terms of their Minimum Detectable Concentration, or MDC. The MDC is the concentration that a given instrument and measurement technique can be expected to detect 95% of the time under actual conditions of use. Instruments and methods used for field measurements generally must be capable of meeting the investigation level such as those shown in Table 5-2. Section 5 of Appendix E to NUREG-1757 (Reference 9) provides a method that can be used to determine instrument sensitivities.

Table 5--2
Final Status Survey Investigation Levels

Survey Unit Classification	For fixed measurements or samples, perform investigation if:	For scan measurements, perform investigation if:
Class 1	> $DCGL_{EMC}$ or > $DCGL_W$ and a statistical outlier.	> $DCGL_{EMC}$
Class 2	> $DCGL_W$	> $DCGL_W$ or > MDC_{scan} if MDC_{scan} is greater than the $DCGL_W$
Class 3	> 0.5 ´ $DCGL_W$	Detectable over background.

5.2.1.2 Scan Coverage Requirements

MARSSIM calls for a graded approach to the requirements for scanning of building surfaces. Table 5-3 gives the area coverage requirements when scanning is used with fixed measurements. As can be seen in this table, when the potential for contamination is higher (e.g., in a Class 1 Area), higher scan coverage is required.

Table 5-3
Traditional Scanning Coverage Requirements

Survey Unit Classification	Required Scanning Coverage Fraction
Class 1	100%

Class 2	Outdoor areas, floors, or lower walls of buildings: 10% to 100% Upper walls or ceilings: 10% to 50%
Class 3	Judgmental

5.2.1.3 Removable Activity

Per MARSSIM, surface contamination DCGLs apply to the total of fixed plus removable surface activity. In the calculation of DCGLs, removable surface contamination was assumed to be 10%. In some cases, smear surveys may be needed to confirm this assumption to be true.

One category of radiological data (e.g., radionuclide concentration, direct radiation level, or surface contamination) may be sufficient to determine the extent of contamination. Other measurements may not be necessary (e.g., removable surface contamination or exposure rate measurements).

5.2.2 Fixed Measurements

Fixed measurements are taken by placing the instrument at the appropriate distance from the surface, taking a discrete measurement for a pre-determined time interval, and recording the reading. Fixed measurements may be collected at random locations in a survey unit or may be collected at systematic locations and supplement scanning surveys for the identification of small areas of elevated activity. Fixed measurements may also be collected at locations identified by scanning surveys as part of an investigation to determine the source of the elevated instrument response. Professional judgment may also be used to identify locations for fixed measurements to further define the areal extent of contamination.

5.2.3 Advanced Technologies

Other instruments and methods that may be used for FSS include, but are not limited to, position-sensitive proportional counters, in situ gamma spectrometry and systems capable of traversing ducting or piping. The purpose of these “advanced technologies” is to increase the efficiency and/or accuracy of the FSS compared to “conventional” survey techniques such as the use of handheld survey instruments by survey personnel. These other methods may in some cases provide sufficient area coverage so that augmenting the measurement with scanning is not necessary. Generally, if advanced technology instrumentation is selected for use as part of an FSS, technical justification is required by the regulators. Alternatively, these types of instruments have been shown to be useful tools during characterization surveys or to determine the effectiveness of remediation activities.

[redacted]

5.2.4 Gross Activity DCGLs

For alpha or beta surface activity measurements, field measurements will typically consist of gross activity assessments rather than radionuclide-specific techniques. Gross activity DCGLs are established, based on the representative radionuclide mix, as follows:

$$DCGL_{GA} = \frac{1}{\sum_{1}^{n} \frac{f_i}{DCGL_i}} \qquad \textbf{Equation 5-1}$$

where: f_i = fraction of the total activity contributed by radionuclide
i = the number of radionuclides
$DCGL_i$ = DCGL for radionuclide i

Gross activity DCGLs can be developed for gross beta measurements, or a gross beta DCGL can be scaled so that it acts as a surrogate for gross alpha (see next section). Equation 5-1 is generally applied for radionuclides that are present in a survey unit in concentrations greater than 10% of their respective DCGL. Per NRC Guidance stated in Section 3.3 of NUREG-1757 (Rev. 2, Reference 14), radionuclides present at less than 10 % of their DCGLs can be considered insignificant (deselected), per NRC guidance, and no longer need to be included in the subsequent FSS for the survey unit. If multiple radionuclides are to be deselected, the total of the percentages of the DCGL for the radionuclides to be deselected can be more than 10 %. Although these deselected radionuclides do not need to be included in the survey design, the total dose for the deselected radionuclides needs to be subtracted from the dose that can be result from the remaining radionuclides. For example, if the dose from the deselected radionuclides is 2.5 mrem/yr (10 % of the 25 mrem/yr limit), the maximum dose allowed from the remaining radionuclides is 22.5 mrem/yr.

5.2.5 Surrogate Ratio DCGLs

Section 2.3.5 illustrated a methodology that can be used to develop the ratios of Hard to Detect (HTD) to Easy to Detect (ETD) radionuclides from site characterization survey results. The following illustrated how these ratios can be used to develop surrogate ratio DCGLs for use in Final Status Surveys.

It is acceptable industry practice to assay a Hard-To-Detect (HTD) radionuclide by using a surrogate relationship to an Easy-To-Detect (ETD) radionuclide. A common example would be to use a beta measurement to assay an alpha emitting radionuclide. Another example would be to relate a specific radionuclide, such as Cesium-137, to one or more radionuclides of similar characteristics. In such cases, to demonstrate compliance with the release criteria for the survey unit the DCGL for the surrogate radionuclide or mix of radionuclides must be scaled to account for the fact that it is being used as an indicator for an additional radionuclide or mix of radionuclides. The result is referred to as the surrogate DCGL.

The following process has been acceptable to the NRC in assessing the need to use surrogate ratios for final status surveys (FSS):

- Determine whether HTD radionuclides (e.g., transuranic radionuclides, Sr-90, H-3) are likely to be present in the survey unit (and in significant quantities as discussed in the last section) based on process knowledge, historical data, or characterization.

- Screen HTD radionuclides using the 10% rule described in Section 2.3.3.
- Radionuclides not screened out will require a surrogate DCGL. Surrogate relationships will be determined from the samples results using one of methods described below.
- Using the ratios developed by a process like that in Section 2.3.5, develop a surrogate DCGL for each HTD radionuclide.

$$DCGL_{surrogate} = DCGL_{ETD}\, x \frac{DCGL_{HTD}}{(f_{HTD\,:\,ETD} \text{ x } DCGL_{ETD}) + DCGL_{HTD}}$$ **Equation 5-2**

Where $DCGL_{surrogate}$ = the DCGL for the easy-to-detect radionuclide after being reduced to reflect the effect of the HTD Radionuclide

$DCGL_{ETD}$ = the DCGL of the easy-to-detect radionuclide prior to adjustment by the surrogate process

$DCGL_{HTD}$ = the DCGL for the hard-to-detect radionuclide being included in the surrogate process

$f_{HTD\,;\,ETD}$ = the activity ratio of the hard-to-detect radionuclide to the easy-to-detect radionuclide

- The radionuclides ratios for a particular survey area may need to be revalidated after remediation has been conducted due to the removal of at least some of the sampled material.

5.2.6 Effect of Hard-To-Detect Radionuclides on Scan Surveys for Structure Surfaces

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For alpha survey instrumentation with a background around one to three counts per minute, a single count will give a surveyor sufficient cause to stop and investigate further. Thus, the probability of detecting given levels of alpha emitting radionuclides can be calculated by use of Poisson summation statistics. Doing so (see MARRSIM Section 6.7.2.2), one finds that the probability of detecting an area of alpha activity of 300 dpm/100cm^2 at a scan rate of 3 cm per second (roughly 1 inch [2.54 cm] per second) is 90% if the probe dimension in the direction of the scan is 10 cm. If the probe dimension in the scan direction is halved to 5 cm, the detection probability is still 70% (Table 6.8 pf MARSSIM). Choosing appropriate values for surveyor efficiency, instrument and surface efficiencies will yield MDCs for alpha surveys for structure surfaces. If for some reason lower MDCs are desired, then scan speeds can be adjusted, within practical limits, via the methods given in Appendix J of the MARSSIM. As alpha scan surveys are generally hard to successfully accomplish, more remediation to avoid the need for separate alpha surveys is generally a more cost-effective strategy.

5.2.7 Additional Building Surface FSS Challenges

5.2.7.1 Condition of Surface to be Surveyed

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5.2.7.2 Reference Areas and Materials

The DQO process, used during the planning phase in the preparation of a final status survey plan, determines whether media specific backgrounds, ambient area background or no background will be applied to a survey area or unit. The approach used for a specific survey unit generally is based on the survey unit classification and the applicable DCGLs.

If applied, media specific backgrounds are determined via measurements made in one or more reference areas and on various materials selected to represent the baseline radiological conditions for the site. The determination of media specific background is controlled with a documented survey plan, which has been prepared using the DQO process. This process will ensure that the collected data meets the needs of the final status survey. The collected data may be used as the reference area data set when using the Wilcoxon Rank Sum test, or, for survey units with multiple materials, background data may be subtracted from survey unit measurements (using paired observations) if the Sign Test is applied.

Table 5-4 gives typical ranges for background radiation (i.e., not resulting from plant operations) emitted from standard construction materials expected to be encountered during final survey activities. Ranges are given for several detector types (gross counters) and encompass the variability expected for different materials. The data in Table 5-4 are derived from both NUREG-1507 (Reference 13) and from experience at the Connecticut Yankee (CY) Plant.

Table 5-4 Typical Media Specific Backgrounds (Reference 13)

Instrument Type	Operating Mode/Model	Nominal Background Range (cpm = counts per minute)
Gas Proportional Counter (100 cm^2)	alpha-only mode	1 cpm - 20 cpm for ceramic tile; 1 cpm - 10 cpm for other materials
	beta-only mode	300 cpm - 1,250 cpm for all materials
	alpha & beta mode	280 cpm - 1,250 cpm for all materials
Pancake GM Probe (20 cm^2)	N/A	40 cpm - 125 cpm for all materials
Zinc Sulfide (100 cm^2)	N/A	1 cpm - 10 cpm for ceramic tile; 1 cpm - 5 cpm for other materials
Plastic Scintillator (100 cm^2)	N/A	500 cpm - 1,500 cpm for all materials
Sodium Iodine (NaI)	1-inch by 1-inch	2,000 cpm - 4,000 cpm for soil
	1.25-inch by 1.25-inch	3,000 - 6,000 cpm for soil
	2-inch by 2-inch	8,000 cpm - 16,000 cpm for soil

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Additional information on how to conduct a background study and the results of work performed at decommissioning sites can be found in NUREG-1507 (Reference 13).

5.2.8 Building FSS Techniques and Alternate Approaches

5.2.8.1 Containment Final Status Survey

Different plants address the survey of the above ground portion of the containment building in different manners. This is due to the height and difficulty in accessing the surfaces of a containment building. The dimensions of the CY Containment Building for example were as follows:

- 170 feet (51.8 m) tall
- 140 feet (42.7 m) in diameter

For the Trojan plant, all interior concrete was removed prior to conducting the final status survey. To access the upper inside areas of the containment building, a very extensive scaffolding structure was constructed on top of the containment crane. As the containment liner was a MARSSIM Class 2, at least 10 % of the liner surface needed to be surveyed. The scaffolding structure allowed access to a portion of the liner at a time. After completion of a section, the crane was rotated to allow survey of another section. One problem with this approach was that a significant amount of additional remediation around the locations of attachments, which had been used to support various components during plant operation, was needed (mostly grinding around weld locations).

The Trojan plant is a good example of a plant that performed Final Status Surveys and was released from its license with essentially all affected buildings in an intact condition. This resulted in more Final Status Surveys being performed than many other plants. The following are the final status survey statistics for the Trojan plant (note that these statistics include surveys of land areas):

- Over 500 FSS units
- 195,000 m^2 of surfaces surveyed
- 60,000 discrete measurements collected

5.2.8.2 Alternate Approaches to Final Status Survey

The information above concerning the surveys of buildings is primarily in conformance with the methods described MARSSIM guidance. The following subsections describe alternate methods that have been approved by the NRC and used at plant sites being decommissioned.

5.2.8.2.1 Basement Fill Model

The calculation performed to determine dose using the Basement Fill Model is based on the total inventory of radioactivity contained in the basement concrete. Concrete samples were used at CY and Maine Yankee to allow determination of this inventory. Scanning for the purpose of performing an FSS was not required at CY but was performed to identify any areas of higher activity that would need additional core samples. In this way the activity inventory determined would be conservative. Details on the basis and use of the Basement Fill Model are provided in chapter 6.

5.2.8.2.2 Use of Soil and Demolition Debris as Backfill

This section will describe experiences at some of the plants that have used soil and/or concrete demolition debris as backfill of excavations or basements that have met the site release criteria.

5.2.8.2.3 Yankee Rowe Plant

Concrete building demolition debris to be used as backfill at the Yankee Rowe Plant had to meet very restrictive criteria imposed by additional requirements placed on the licensee by the Massachusetts Department of Environmental Protection (MassDEP). To comply with this requirement, Yankee Rowe developed a special technique to determine that material to be used as backfill was "free releasable." The concrete demolition debris and other backfill materials were placed in large dump trucks. Yankee Rowe purchased a gamma spectroscopy system consisting of a total of 6 detectors and associated computers placed in two trailers. The detectors were mounted on vertical rails so that the concrete loaded in the trucks dump bed could be more completely analyzed. A similar system was used during the decommissioning of the Big Rock Point and Humboldt Bay plants to survey concrete debris and soil for use as backfill of excavations and on site.

5.2.8.2.4 Zion Plant – Use of soil as backfill

The experiences from the Zion plant in this chapter are from the Zion LTP (Reference 15). The Zion plant did not stockpile excavated soil for reuse as backfill in basements. However, overburden soil was excavated to expose buried components (e.g., concrete pads, buried pipe, buried conduit, etc.) for removal and disposal or to allow the installation of a new buried system. In these cases, the overburden soil was replaced back into the excavation after the completion of a Radiological Assessment (RA).

In these cases, when an RA is conducted, the following steps were taken:

- The footprint of the excavation and areas adjacent to the excavation where the soil was to be staged were scanned prior to the excavation.
- Periodic scans were performed on the soil as it was excavated.
- Exposed surfaces of the excavated soil pile were scanned prior to reuse as excavation backfill.
- Scanning was performed using NaI gamma scintillation detectors (typically 2" x 2"). When using hand-held detectors, gamma scanning is generally performed by moving the detector in a serpentine pattern, usually within 15 cm (6 in) from the surface, while advancing at a rate of approximately 0.5 m (20 in) per second. Audible and visual signals are monitored by surveyors who respond to indications of elevated areas while surveying. Upon detecting an increase in visual or audible response, the surveyor reduced the scan speed or pause and attempted to isolate the elevated area. If the elevated activity is verified the location is investigated with a soil sample as described in the next bullet.

- Soil samples were acquired at any scan location that indicated activity of more than 50% of the soil Operational DCGL (See Appendix B for general discussion of operational DCGLs). Any soil confirmed as containing residual radioactivity at concentrations exceeding 50% of the soil Operational DCGL were not used to backfill the excavation and were disposed of as waste.

An RA was also performed at Zion prior to introducing off-site material for use as backfill in a basement, or for any other use. The RA was performed at the borrow pit, landfill, or other location from where the material originated and will consist of gamma scans and material sampling as described above. Gamma scans are performed in situ, or by package (using a hand-held instrument or truck monitor). Soil samples of overburden soils will be analyzed by gamma spectroscopy.

5.2.8.2.5 Zion Plant – Use of Concrete Demolition Debris as Backfill

As background to the use of concrete as backfill at Zion, an explanation of the use of the Basement Fill Model (BFM) at Zion is first needed.

The BFM source term for a given basement structure includes the contributions from basement surfaces, (concrete and steel liner for containment), embedded pipe and penetrations that are contained in, or interface with, the basement. Each dose component (surface, embedded pipe, penetrations) has a unique DCGL. Concrete fill is another dose component applicable to any basement where clean concrete debris is used as fill.

The total dose attributed to the use of concrete fill for each basement, including all Radionuclides of Concern (ROC) is dependent on which structure's basement is being backfilled, and are listed below. The dose values (in mrem/yr) are initial conservative values to be added to any basement where concrete fill was used regardless of the volume of concrete fill used.

- Auxiliary Building 0.99
- Containment 1.77
- SFP/Transfer Canal 0.15
- Turbine Building 1.58
- Crib House/Forebay 1.57
- Wastewater Treatment Facility (WWTF) 6.40

The *a priori* doses from clean concrete fill provided above are based on the maximum allowable minimum detectable concentration (MDC) of 5,000 dpm/100cm^2, which corresponds to the survey instrument sensitivity. Assigning dose to this concentration is a conservative assumption. This is solely a bounding value and not indicative of the actual MDC values experienced when conducting the Unconditional Release Surveys (URS) performed on the concrete, which were significantly lower. After all URS were completed on the remainder of the concrete to be reused as clean fill, the dose from fill given above were recalculated based on the actual maximum MDC observed during the performance of the URS.

After the FSS of all dose components in each basement were complete and all dose component survey units pass the Sign Test, the SOF for each dose component is calculated by dividing the basement-specific assigned dose such as those provided above by 25 mrem/yr.

5.2.9 Survey of Non-RCA Buildings

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5.2.10 Survey Protocol for Non-Structural Systems and Components

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5.2.10.1 Zion FSS Experience with Embedded Piping

As discussed in section 5.2.8.2.1, a contribution to the total BFM dose needed to be calculated for embedded piping at Zion. The BFM groundwater source term transport and dose assessment pathways applicable to embedded pipe are the same as those assumed for concrete. In this approach, the activity in the pipe is assumed to be instantaneously released and mixed with the water in the interstitial spaces of the fill material with the water then used for drinking and irrigation.

An FSS was conducted on the interior surfaces of embedded piping to demonstrate that the concentrations of residual radioactivity are equal to or below DCGLs corresponding to the dose criterion in 10 CFR 20.1402 ($DCGL_{EP}$). $DCGL_{EP}$ were calculated for each of the embedded pipe survey units. The $DCGL_{EP}$ values from Zion LTP are reproduced in the table below. The Inconsequential Contaminants (IC) dose percentages of 10% for containment and 5% for all other survey units were used to adjust the $DCGL_{EP}$ values in the table to account for the dose from the eliminated IC radionuclides.

Table 5-4
Base Case DCGLs for Embedded Pipe ($DCGL_{EP}$)

Radionuclide	**Auxiliary Bldg. Basement Embedded Floor Drains (pCi/m^2)**	**Turbine Bldg. Basement Embedded Floor Drains (pCi/m^2)**	**Unit 1 & Unit 2 Containment In-Core Sump Embedded Drainpipe (pCi/m^2)**	**Unit 1 & Unit 2 Steam Tunnel Embedded Floor Drains (pCi/m^2)**	**Unit 1 & Unit 2 Tendon Tunnel Embedded Floor Drains (pCi/m^2)**
H-3	N/A	N/A	N/A	N/A	N/A
Co-60	7.33E+09	6.31E+09	5.47E+09	4.07E+10	1.06E+10
Ni-63	2.78E+11	1.96E+11	1.40E+11	1.26E+12	2.72E+11
Sr-90	2.41E+08	6.94E+07	4.98E+07	4.48E+08	9.70E+07
Cs-134	5.10E+09	1.43E+09	1.05E+09	9.22E+09	2.04E+09
Cs-137	2.68E+09	1.89E+09	1.37E+09	1.22E+10	2.67E+09
Eu-152	N/A	N/A	1.28E+10	N/A	2.48E+10
Eu-154	N/A	N/A	1.11E+10	N/A	2.16E+10

5.2.10.2 Zion FSS experience with penetrations

A penetration is defined for the Zion FSS as a pipe (or remaining pipe sleeve, if the pipe is removed, or concrete, if the pipe and pipe sleeve is removed) that runs through a concrete wall and/or floor, between two buildings, and is open at the wall or floor surface of each building. A penetration could also be a pipe that runs through a concrete wall and/or floor and opens to a building on one end and the outside ground on the other end.

A penetration survey unit is defined for each basement. The direction that the residual radioactivity may migrate, i.e., into which basement, cannot be predicted with certainty. Therefore, a given penetration that begins in one basement and ends in another will be included in the survey units for both basements. The residual radioactivity in the penetration is assumed to be released to both basements simultaneously.

The BFM groundwater source term transport and dose assessment pathways applicable to penetrations are the same as those assumed for concrete, i.e., the activity in the penetration is released and mixed with the water in the interstitial spaces of the fill material with the water then used for drinking and irrigation. Note that the DCGLs calculated for penetrations are based on an assumption of instant release of all activity into the basement fill.

An FSS was conducted at Zion on the interior surfaces of penetrations to demonstrate that the concentrations of residual radioactivity are equal to or below DCGLs corresponding to the dose criterion in 10 CFR 20.1402 ($DCGL_{PN}$). By definition, a given penetration interfaces two basements. The lesser DCGLPN of the two basements were used for remediation and grouting action levels. $DCGL_{PN}$ were calculated for each of the embedded pipe survey units. The DCGLPN values from Zion LTP are reproduced in Table 5-5 (referred to as Base Case DCGLs for penetrations in this Chapter). The IC dose percentages of 10% for Containment and 5% for all other survey units were used to adjust the DCGLPN values in Table 5-5 to account for the dose from the eliminated IC radionuclides.

Table 5-5
Base Case DCGLs for Penetrations ($DCGL_{PN}$)

Radionuclide	Auxiliary Bldg. (pCi/m^2)	U1/U2 Containment (pCi/m^2)	SFP/Transfer Canal (pCi/m^2)	Turbine Bldg. (pCi/m^2)	Crib House/ Forebay (pCi/m^2)	WWTF (pCi/m^2)
H-3	3.99E+09	3.42E+09	4.84E+16	3.23E+09	N/A	N/A
Co-60	8.82E+07	2.26E+09	4.45E+08	1.76E+09	N/A	N/A
Ni-63	6.79E+10	5.78E+10	1.86E+14	5.48E+10	N/A	N/A
Sr-90	2.41E+07	2.06E+07	9.26E+10	1.94E+07	N/A	N/A
Cs-134	3.28E+08	4.32E+08	7.48E+08	4.00E+08	N/A	N/A
Cs-137	6.17E+08	5.66E+08	1.46E+09	5.29E+08	N/A	N/A
Eu-152	3.29E+08	5.26E+09	9.44E+08	4.06E+09	N/A	N/A
Eu-154	2.33E+08	4.58E+09	8.53E+08	3.58E+09	N/A	N/A

5.3 Survey Considerations for Outdoor Areas

5.3.1 Residual Radioactivity in Surface Soils

In this context, surface soil refers to outdoor areas where the soil is, for purposes of dose modeling, considered to be uniformly contaminated from the surface down to some specified depth [usually 6 inches (15cm)] (Reference 14, Section 3.6). These areas can be surveyed through combinations of sampling, scanning, and in-situ measurements, as appropriate.

The general approach prescribed by MARSSIM for final status surveys requires that at least some minimum number of measurements or samples be taken within a survey unit, so that the non-parametric statistical tests used for data assessment can be applied with adequate confidence. Decisions regarding whether a given survey unit meets the applicable release criterion are made based on the results of these tests. Scanning measurements are used to check the design basis for the survey by evaluating if any small areas of elevated activity exist that would require reclassification, tighter grid spacing for the fixed measurements, or both. The conventional method of performing a scan survey for a land area involves the use of a handheld sodium iodide detector operated by survey technicians. The technician walks slowly across the survey unit while swinging the detector slowly back and forth. If the detector increases in meter response the technician stops to confirm the increase. If confirmed, the location of the increased reading is marked for further investigation which may include taking of a soil sample. As this method is discussed in MARSSIM it will not be covered in further detail in this report.

For combinations of fixed measurements and traditional scanning, the MARSSIM methodology is to select a requisite number of measurement locations to satisfy the decision error rates for the non-parametric statistical test to be used for data evaluation and to account for sample losses or data anomalies. The purpose of scans is to confirm that the area was properly classified and that any small areas of elevated activity are within acceptable levels (i.e., are less than the applicable $DCGL_{EMC}$). Depending on the sensitivity of the scanning method used, the number of fixed measurement locations may need to be increased so the spacing between measurements is reduced. Under MARSSIM, the level of survey effort required for a given survey unit is determined by the potential for contamination as indicated by its classification. Typical coverage requirements that are applied for scans performed in support of final status surveys of land areas are:

- For Class 1 survey units, 100% of the area will be scanned.
- For Class 2 survey units, between 10% and 100% of the area will be scanned in a combination of systematic and judgmental measurements.
- Scanning will be done on a judgmental basis for Class 3 survey units.

Some details concerning the scanning of land areas are included in section 5.2.8.2.4.

5.3.1.1 Advanced Technology

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5.3.1.2 Fixed Measurement Requirements

For fixed measurements, MARSSIM states that MDCs should be as far below the $DCGL_W$ as possible, with values less than 10% of the $DCGL_W$ being preferred, and up to 50% of the $DCGL_W$ being acceptable. Note that “fixed measurements” is used interchangeably to refer to measurements or samples taken at specific locations. If the advanced technology has sufficient sensitivity (i.e., low enough MDC) it may satisfy both the scanning and fixed measurement MDCs in one step. In-situ gamma spectroscopy and SCM discussed in Appendix A are examples of advanced technologies that may have this capability.

- For Class 1 or Class 2 survey units, advanced survey technologies may be used exclusively only in survey units for which the above coverage requirements can be achieved and MDCs are no greater than 50% of the applicable $DCGL_W$.

- For Class 1 or Class 2 survey units for which advanced technologies would have an acceptable MDC, but the above coverage requirements cannot be achieved, advanced technologies may be used over 100% of the accessible area with a combination of fixed measurements and traditional scans used over the remainder of the area.

5.3.1.3 Background Reference Area Determination

The MARSSIM process allows the subtraction of background for radioactivity present in the soil at the site due to fallout from events such as nuclear weapons testing. Generally, the radionuclides present in fallout are Cs-137 and Sr-90 at a lesser concentration. Determining the background concentration involves identifying a suitable area (generally off-site) that has not been affected by releases from the site. Identifying an area that adequately represents the plant site in terms of topography, soil type and disturbance history has required a high degree of interaction with the regulators. Due to the difficulty of performing the work to justify background reference areas for the use in determining values for background in soil and considering the average background concentrations are in the range of 5 to 10 % of the Cs-137 soil screening DCGLs most plants have not attempted to justify a background value(s) for subtraction during the FSS process.

Much of the above discussion concerning the FSS of surface soils has been kept at a high level as the details of how to conduct this type of survey are covered extensively in MARSSIM. It is the primary intent of this report to focus on information not covered in regulatory guidance documents.

5.3.2 Residual Radioactivity in Subsurface Soil

Although MARSSIM has clear guidance on the conduct of a Final Status Survey for surface soil in land areas, it does not provide guidance on Final Status Surveys for land areas that have subsurface contamination. As Connecticut Yankee (CY) and Humboldt Bay Power Plant (HBPP) were sites that exhibited subsurface contamination, the experience at those plants in addressing this issue is presented in the following subsections.

5.3.2.1 Connecticut Yankee Subsurface Soil FSS

The soil DCGLs developed for a typical site assumes that contamination is present only in the surface soil (i.e., to a depth of 6 inches (15 cm)). Due to extensive subsurface contamination present on site, the soil DCGLs for CY were developed assuming that the contamination was homogeneous over a depth of 3 meters. The limits developed by this model were therefore conservatively applied at CY to both the surface FSS and the subsurface FSS (applied over the first 3 meters of soil). As approved by the NRC via the CY License Termination Plan, the subsurface sampling was conducted to match this dose modeling scenario. A direct push sampler was utilized to collect soil cores approximately 1 meter in length.

Samples were obtained to a depth of 3 meters or bedrock, whichever is reached first. These samples were then homogenized over the entire depth of the sample obtained. In cases where probe refusal was met because of bedrock, the sample was used “as is.” In cases where a non-bedrock refusal was met prior to the 3-meter depth, the available sample was used to represent the 3-meter sample if the viable sample was at least 1.5 meters in depth. If a non-bedrock refusal was met before the 1.5-meter depth, then a new sample was obtained within a 3-meter radius from the original location. All samples were analyzed by gamma spectrometry. Because of the mobility of some of the HTD radionuclides in groundwater, some of the samples underwent analysis for all HTD radionuclides. A minimum of 5% of the samples were analyzed for HTD radionuclides. During specific investigations, such as the identification of the horizontal extent of contamination, analysis of a larger percentage of samples for HTD radionuclides was performed.

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Another aspect of the subsurface FSS that needed to be developed was the sample density. NRC approved a graded approach where the survey density was highest in areas with the highest potential for subsurface contamination. Class designations were assigned to the different subsurface areas. The total area of the site for which subsurface surveys were conducted was approximately 25 acres (10.1 hectares) or approximately 5% of the total site area of 525 acres (213 hectares). Class A areas have the highest survey density while Class C had the lowest (See Table 5-6). The survey area corresponding to the former Radiological Control Area was classified as Class A while some of the areas surrounding the industrial area were classified as Class B or C due to the lower potential for subsurface contamination.

Table 5-4
CY Subsurface Soil Sampling Density

Subsurface Classification of Survey Unit	Potential for Subsurface Contamination	Number of Samples per Survey Unit
A	High	31 (1 per each 500 m^2)
B	Medium	25
C	Low	15

In addition, biased samples (taken from locations suspected of containing higher levels of contamination) were obtained at the locations of localized remediation efforts where subsurface leaks were suspected to have contaminated the subsurface soil. Random samples were obtained in Class B and Class C areas. Biased samples were obtained in Class B and Class C areas based upon characterization data and professional judgment. If a systematic or random sample location fell on a building foundation, a sample was obtained at that location unless the building was in contact with bedrock. The range of the number of measurements in Class A, B, and C areas (31 measurements in the Class A area to 15 measurements in Class C areas) corresponds to the typical range of values for the number of samples required per MARSSIM defined protocols. All samples were evaluated against the soil DCGLs by using the Sign test.

Investigation levels applicable to surface soils were applied to subsurface soils (see Table 5-1). Similarly, the area factors for surface soils were applied to subsurface soils, i.e., no sample can exceed the $DCGL_{EMC}$ without an investigation being performed. These investigations were like those performed for surface soils.

The horizontal extent of contamination was only established for judgmental sampling and for samples within a systematic sampling area that exceeds the $DCGL_{EMC}$. For the case where the $DCGL_{EMC}$ comparison was made, the value used for the area factor was determined from the area bounded by the adjacent samples or by the area bounded by additional samples at or below the $DCGL_{W}$. This approach is consistent with the model used to calculate DCGLs in Section 6.

5.3.2.2 Zion Subsurface Soil FSS

Subsurface soil refers to soil that resides at a depth greater than 15 cm below the final configuration of the ground surface or soil that remained beneath structures such as basement floors/foundations or pavement at the time of license termination.

Any soil excavation created to expose or remove a potentially contaminated subgrade basement structure was subjected to FSS prior to backfill. The FSS was designed as an open land survey using the classification of the removed structure in accordance with the Zion LTP using the Operational DCGLs for subsurface soils as the release criteria.

During decommissioning of Zion, any subsurface soil contamination that was identified by continuing characterization or operational radiological surveys that exceed the site-specific Base Case DCGLs for each of the potential Radionuclides of Concern (ROC) were remediated. The remediation process included performing Remedial Action Support Survey (RASS as defined in MARSSIM) of the open excavations as summarized in the following.

The RASS included scan surveys and the collection of soil samples during excavation to gauge the effectiveness of remediation, and to identify locations requiring additional excavation. The scan surveys and the collection of and subsequent laboratory analysis of soil samples were performed in a manner that is intended to meet the DQOs of FSS. The data obtained during the RASS is expected to provide a high degree of confidence that the excavation, or portion of the excavation, meets the criterion for the unrestricted release of open land survey units. Soil samples were collected to depths at which there is high confidence that deeper samples did not result in higher concentrations.

Alternatively, an NaI detector or intrinsic germanium detector of sufficient sensitivity to detect residual radioactivity at the Operational DCGL could be used to scan the exposed soils in an open excavation to identify the presence or absence of soil contamination, and the extent of such contamination. If the detector identifies the presence of contamination at a significant fraction of the Operational DCGL, additional confirmatory investigation and analyses of soil samples of the suspect areas were performed.

Scanning of Subsurface Soils during FSS

Per NUREG-1757, scanning is not applicable to subsurface soils during the performance of FSS. Scanning was performed during the RASS of excavations resulting from any remediation of subsurface soil contamination. The scanning of exposed subsurface soils during the RASS, when accessible as an excavated surface, was used with the analysis of soil samples to demonstrate compliance with site release criteria.

Sampling of Subsurface Soils during FSS

In accordance with NUREG-1757, Volume 2, Revision 1, Appendix G, if the HSA indicates that there is no likelihood of substantial subsurface residual radioactivity, subsurface surveys are not necessary. The HSA as well as the results of the extensive characterization of subsurface soils in the impacted area surrounding the Zion facility had shown that there is minimal residual radioactivity in subsurface soil. Consequently, Zion proposed to perform minimal subsurface sampling during FSS.

In Class 1 open land survey units, subsurface soil samples were taken at 10% of the systematic surface soil sample locations in the survey unit with the location(s) selected at random. In addition, if during the performance of FSS, the analysis of a surface soil sample, or the results of a surface gamma scan indicated the potential presence of residual radioactivity at a concentration of 75% of the subsurface Operational DCGL, then additional biased subsurface soil sample(s) were taken within the area of concern as part of the investigation.

In Class 2 and Class 3 open land survey units, no subsurface soil sample(s) were taken as part of the survey design. However, as with the Class 1 open land survey units, if during the performance of FSS, the analysis of a surface soil sample, or the results of a surface gamma scan indicated the potential presence of residual radioactivity at a concentration of 75% of the subsurface Operational DCGL, then biased subsurface soil sample(s) were taken to the appropriate depth within the area of concern as part of the investigation.

GeoProbe®, split spoon sampling or other methods can be used to acquire subsurface soil samples. Subsurface soil samples were obtained to a depth of at least 1 meter or refusal, whichever was reached first. In cases where refusal was met because of bedrock, the sample was used "as is." In cases where a non-bedrock refusal was met prior to the 1-meter depth, the available sample was used to represent the 1-meter sample. If residual radioactivity was detected in the 1-meter sample, an additional meter of depth was sampled and analyzed.

Subsurface soil samples were segmented and homogenized over each one meter of depth. Extraneous material was removed from each segment and the sample was adequately dried. The material was then placed into a clean sample container and properly labeled. All samples were tracked from time of collection through the final analysis in accordance with procedure and survey package instructions.

All subsurface soil samples taken during continuing characterization and FSS were analyzed by gamma spectrometry.

Sampling of Subsurface Soils Below Structure Basement Foundations

The foundation walls and basement floors below the 588-foot elevation of the Unit 1 Containment, Unit 2 Containment, Auxiliary Building, Turbine Building, Crib House/Forebay, WWTF and remnants of the SFP remained at the time of license termination. Based on the results of subsurface soil sampling performed during site characterization; it was not likely that the residual radioactivity concentrations in soil beneath these building foundations exceeded the site-specific Base Case DCGLs as presented in Table 5-6. However, prior to license termination, it was necessary to ascertain the radiological conditions of these sub-slab soils to demonstrate suitability for unrestricted release.

The soils under the basement concrete of the Containment Buildings, the Auxiliary Building and the SFP/Transfer Canal were designated as "continuing characterization" areas once commodity removal and building demolition had progressed to a point where access could be achieved. Continuing characterization consisted of soil borings or use of GeoProbe® technology at the nearest locations along the foundation walls that could be accessed. The under-basement soil activity was determined by interpreting results from borings collected at the nearest locations. Locations selected for sampling were biased to locations having a high potential for the accumulation and migration of radioactive contamination to sub-surface soil. Angled soil borings were also performed to directly access the sub-slab soils. The exact number and location of the soil borings were determined by DQO during the survey design. All samples taken from sub-slab soils were analyzed by gamma spectrometry. Ten percent (10%) of any sub slab soil samples taken were analyzed for the initial suite of HTD radionuclides as well as for any individual sample where analysis indicated gamma activity in excess of a sum of fractions (SOF) of 0.1.

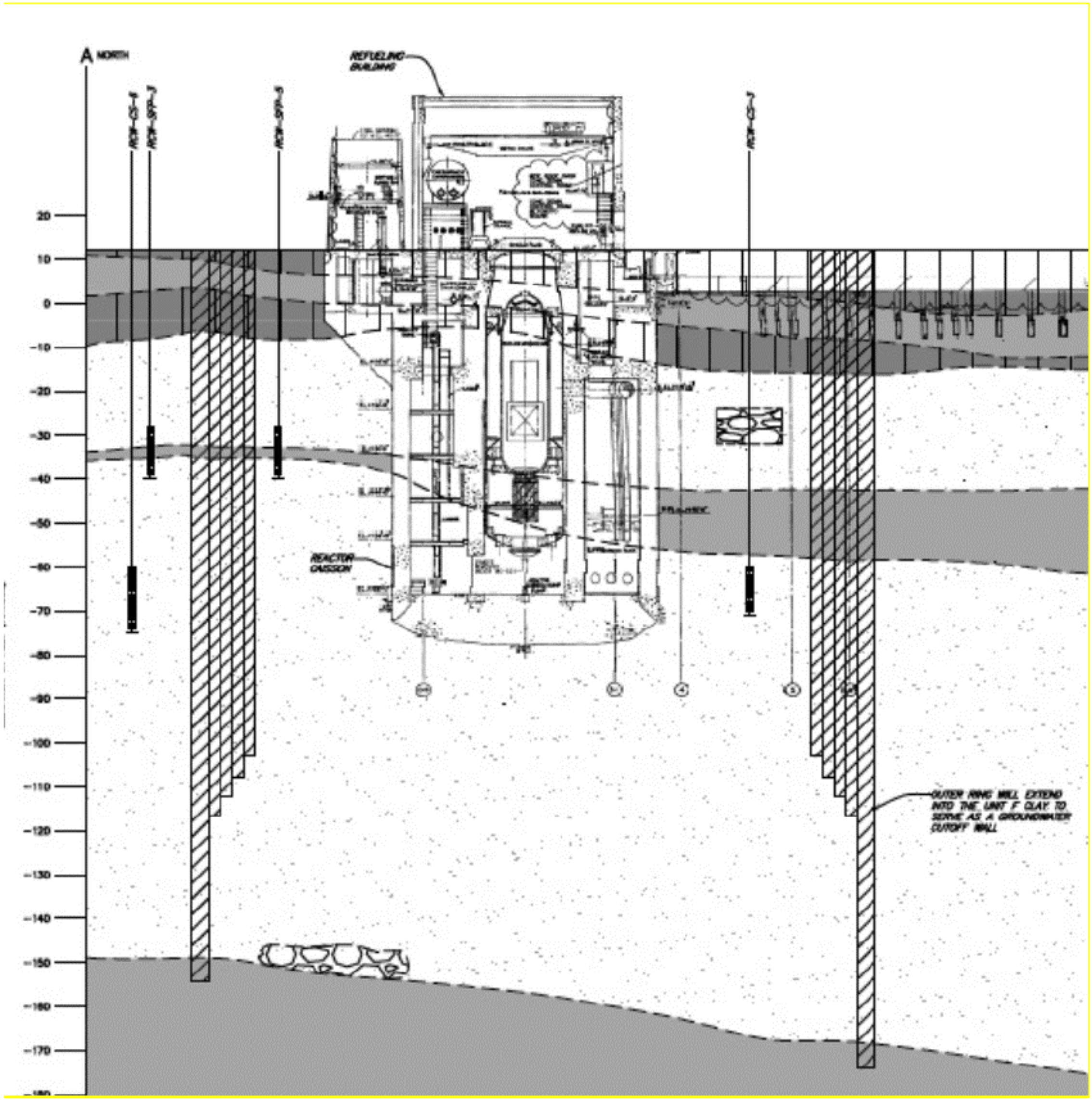
Where possible, survey design considered the possibility of coring through the basement concrete floor slabs to facilitate the collection of soil samples. However, this was not possible due to the intrusion of groundwater into the basements through the bore hole. This was especially true for the containment basements, as sampling through the foundation would require compromising the integrity of the internal steel liner. To address the issue of groundwater intrusion and still investigate the potential for migration of contamination from building interiors to the sub-foundation soils, any continuing characterization performed in the auxiliary building basement, under-vessel area of the containments and the SFP/transfer canal included cores into the concrete floor, but not fully through the foundation or liner. The cores were biased to areas with higher potential of providing a pathway for migration of contamination to sub-foundation soil including stress cracks, floor, and wall interfaces, and penetrations through walls and floors for piping. If the analysis of the deepest 0.5 inch “puck” from the core in the foundation did not contain detectable activity, then it was assumed that the location was not a source of sub-foundation soil contamination. If activity was positively detected at the deepest point in the core, continuing the core to the soil under the foundation was considered depending on the levels of activity identified and the potential for groundwater intrusion.

If residual radioactivity was detected in subsurface soils adjacent to or under a basement surface, then the investigation also included an assessment of the potential contamination of the exterior of the structure. A sample plan for the investigation was created as specified by procedure and the plan and investigation results were provided to NRC for evaluation. Based on the results of the investigation, Zion assessed the dose consequences of the subsurface soil contamination or were remediated as necessary.

5.3.2.3 FSS of Caisson Area at Humboldt Bay

A caisson is a water-tight structure typically used as a foundation, or to carry out work underwater. In the case of HBPP Unit 3, the caisson was a first-of-its-kind structure to house a nuclear containment structure, pressure suppression chamber, bio-shield wall surrounding an RPV and nuclear steam supply system below grade. Due to a number of factors, it was not practical to decontaminate the Humboldt Bay caisson so that it would meet site release criteria. Special steps needed to be taken to remove the caisson due to the high-water table at the site. These steps included the installation of a Cutter Soil Mix (CSM) wall around the caisson to cut off the migration of groundwater and allow dewatering of the area. Figure 5-1 show the caisson and with the CSM in place around it.

Figure 5-1
CSM Wall Vertical Configuration



Due to the method used to install the CSM the typical surface soil and standing structure dose models could not be used. The following describes an unusual specific dose assessment that was performed for the area of the CSM wall. The additional dose calculated using this specific model was added to other doses calculated for the area inside the CSM and the result compared to the site release criteria.

Because the CSM wall would need to be left in place once the caisson was excavated and the resulting hole backfilled, the impact from past spent fuel pool leakage on the soil to be incorporated into the concrete of the CSM wall had to be assessed. Based on the hydrologic site model and past observed monitoring well data, a biased sampling plan of subsurface soil in the path of the CSM wall was developed. Eleven 80-foot soil borings were collected from the soil in the area that the CSM wall would be located. Composite samples from this deep sampling were analyzed to provide an average concentration of plant related radionuclides to be incorporated into the wall. Since approximately 50% of the soil was replaced by cement, these obtained values, when compared to the soil DCGLs for the site yielded a conservative value for the dose from the CSM material. The highest soil concentration measured was 22% of the Humboldt Bay Soil DCGLs with the average concentration of the eleven samples being 6% of the DCGLs.

Four 126-foot-deep dewatering wells located inside the deep shoring system allowed for dewatering, providing for dry excavation of deep structures. Due to their depth of installation, approximately 30 feet below the bottom of the excavation, the well casings and gravel packs in the lower portion of the wells were to be left in place once closed in accordance with state requirements.

The one well in the northeast corner early in the caisson excavation had its pump fail early in the caisson work. As a result, the decision was made to use this well to dispose of accumulating water from misting and rainfall during the excavation. This water was allowed to flow to this well casing hole. The water would then flow through the gravel pack to the soil and ultimately be pumped from the excavation by the other three dewatering wells.

The fines in the water to the northeast well contained both contaminated soil as well as activated concrete fines produced from demolition of the drywell bioshield around the reactor. The gravel pack around the bottom of the well casing designed to keep out soil fines worked in the opposite to capture the soil and concrete fines to the point that the gravel packs and the bottom 10 feet or more of the well casing were "plugged" with radioactively contaminated fines.

The decision was made to clean the "sludge" from above the pump within the casing by sluicing it to capture boxes for dewatering, pull the pump and do final cleaning of the well casing. The average of two composite samples pulled from the sludge were more than three times the soil DCGLs. With the inside of the well casing cleaned, the question became the status of the surrounding gravel packs.

Gross gamma measurements in µr/hr were performed at 1-foot incremental readings down the well casing using a 3" x 3" sodium iodide detector. Utilizing the gamma signature for the samples and assumed densities for the well casing and pea gravel/soil/concrete mixture around the casing, the remaining activity in the gravel pack was analyzed using the Microshield© computer code to determine the concentration in the surrounding soil. The analysis conclusion was that the contamination in the gravel pack around the well casing exceeded the soil DCGL, however, was 31% of the Elevated Measurement Comparison DCGL and therefore was acceptable to remain in the gravel pack without further remediation effort.

5.3.3 Paved Areas

Paved areas that remained following decommissioning activities could require surveys for residual radioactivity on the surface, beneath the surface, or both. As part of the survey design and planning process, historical information can be reviewed to determine whether radiological incidents or plant alterations have occurred in the survey unit. If mixing of impacted soil could have occurred in preparing an area for paving, the mixing depth needs to be considered in the Final Status Survey design of the area. If it is determined that the soil beneath pavement has been impacted, the final status survey needs to incorporate appropriate surveys and sampling.

If residual radioactivity is primarily on or near the surface of the paved area, an acceptable survey method is to take measurements as if the area were surface soil. If the residual radioactivity is primarily beneath the paving, it can be treated, for purposes of surveying, as subsurface residual radioactivity.

5.3.4 Groundwater Assessments

A dose to a future user of the site due to the presence of radionuclides in groundwater must also be included with the dose from other pathways in showing compliance with the site release regulations. The NEI document, NEI 07-07, Rev. 1 (Reference 16) provides guidance as to the methods to characterize, remediate (if needed) and monitor the effectiveness of any remediations towards showing this compliance. The EPRI Groundwater and Soil Remediation Guidelines (Reference 3) provides detailed guidance as to how to meet the guidance statement of NEI 07-07.

Appendix G contains an example for characterization, remediation and final assessment of groundwater contamination at one site.

5.3.5 Bedrock Assessments

[redacted]

[redacted]

- [redacted]
- [redacted]
- [redacted]

[redacted]

[redacted]

[redacted]

[redacted]

5.3.6 Storm Drains and Other Buried Piping

Any storm drains or other buried piping to remain on a site at the time of license termination must be surveyed to the limits such as those given in chapter 6 for buried pipe, or another NRC approved approach.

5.3.7 Final Status Survey and/or Radiological Assessment of Excavations

The Final Status Survey of excavations has been conducted in a few different methods at power plant sites. Two approaches that have been used are the following:

- The surfaces of the excavation are treated as a surface survey unit and surveyed to standard MARSSIM FSS methodology.
- The surfaces of bedrock and soil were assessed as described above for Connecticut Yankee

[redacted]

[redacted]

Connecticut Yankee worked with the Connecticut Department of Environmental Protection (CT DEP) to obtain a change to the site NPDES permit. This change permitted the discharge of groundwater from extraction wells used to dewater the excavation area and from excavations created by remediation activities. The permit change allowed continuous discharge of groundwater after filtration through a portable filtration system capable of up to 450 gpm. Monitoring of the water for pH was also required and this portable filtration system could adjust pH continuously.

To obtain the permit change, monitoring well sample data was needed to show that discharge of the groundwater without radionuclide removal would be within plant effluent limits. Once dewatering wells were installed, a demonstration was required by the regulators to confirm the well sample data. This demonstration involved the pumping of water from the dewatering wells into a portable tank. A sample was withdrawn from the tank and the analysis results compared to effluent limits. The demonstration results confirmed the earlier predictions, and the permit revision was approved. The project was very successful, allowing sufficient remediation to bring the site groundwater levels below the limits for NRC License Termination at a reasonable cost and very low waste generation (approximately 700 ft^3 or 19.8 $m^3$of bag filters, carbon filter media and removed solids).

5.4 Survey Data Assessment

Once measurement data is collected by an FSS, the quality of the data needs to be determined. The information in this section is typically contained in the plant's Quality Assurance Plan (QAP).

The Data Quality Assessment (DQA) process is an evaluation method typically used during the assessment phase of FSS to ensure the validity of FSS results and demonstrate achievement of the survey plan objectives. The level of effort expended during the DQA process will typically be consistent with the graded approach used during the DQO process. The DQA process typically includes:

- A review of the DQOs and survey plan design
- A review of preliminary data
- Use of an appropriate statistical testing when applicable (statistical testing is not always required, e.g., when all sample or measurement results are less than the $DCGL_W$)
- Verification of the assumptions of the statistical tests
- Developing conclusions from the data

Prior to evaluating the data collected from a survey unit against the release criterion, the data are first confirmed to have been acquired in accordance with all applicable procedures and QA/QC requirements. Any discrepancies between the data quality or the data collection process and the applicable requirements are resolved and documented prior to proceeding with data analysis. Data assessment needs to be performed by trained personnel, using approved procedures.

The first step in the data assessment process is to convert all the survey results to DCGL units. Next, the individual measurements and sample concentrations are compared to DCGL levels for evidence of small areas of elevated activity or results that are statistical outliers relative to the rest of the measurements. Graphical analyses of survey data that depict the spatial correlation of the measurements are especially useful for such assessments. The results may indicate that additional data or additional remediation and resurvey is necessary. If this is not the case, the survey results are then evaluated using direct comparisons or statistical methods, as appropriate, to determine if they exceed the release criterion. If the release criterion has been exceeded or if results indicate the need for additional data points, appropriate further actions are then determined.

In some cases, the data set may show radioactivity results that appear to be outliers when compared to adjacent samples. These types of results should be evaluated thoroughly to determine if these results could be a result of other anomalies or errors such as:

- Cross contamination with other samples in the field or within a laboratory environment
- Incorrect sample collection date rendering automatic decay correction in error
- Gamma ray peak misidentification or other analytical interferences
- Sample misidentification

- Elevated concentrations from unknown events rendering the result as inaccurate

In some cases, such potentially anomalous data may represent statistical outliers, but these cases should be rare and not designated as an outlier without due consideration of the above potential causes or others that are not listed, as this list is not necessarily all inclusive.

Interpreting the results from a survey is most straightforward when all measurements are higher or lower than the $DCGL_W$. In such cases, the decision that a survey unit meets or exceeds the release criterion requires little in terms of data analysis. However, formal statistical tests provide a valuable tool when a survey unit's measurements are neither clearly above nor entirely below the $DCGL_W$.

If the survey data are in the form of gross (non-radionuclide-specific) measurements or if the radionuclide of interest is present in background in a concentration that is a relevant fraction of the $DCGL_W$, then the initial data evaluation as described in Table 5-5 can be used. See section 5.3 for discussion of Background Reference Areas that is applicable to Tables 5-5 and 5-6.

Table 5-5
Initial Evaluation of Final Status Survey Results (Background Reference Area Used)

Evaluation Result	Conclusion
Difference between the maximum concentration measurement for the survey unit and the minimum reference area concentration is less than the $DCGL_W$	Survey unit meets the release criterion
Difference between the average concentration measured for the survey unit and the average reference concentration is greater than the $DCGL_W$	Survey unit does not meet the release criterion
Difference between any individual survey result and any individual reference area concentration is greater than the $DCGL_W$ and the difference between the average concentration and the average for the reference area is less than the $DCGL_W$	Conduct either the Wilcoxon Rank Sum test or the Sign test; and the EMC test

If the survey data are in the form of radionuclide-specific measurements and the radionuclide(s) of interest is not present in background in a concentration that is a relevant fraction of the $DCGL_W$, then the initial data evaluation described in Table 5-6 can be used.

Table 5-6
Initial Evaluation of Final Status Survey Results (Background Reference Area Not Used)

Evaluation Result	Conclusion
All measured concentrations less than the $DCGL_W$	Survey unit meets the release criterion

Average concentration exceeds the $DCGL_W$	Survey unit does not meet the release criterion
Individual measurement result(s) exceeds the $DCGL_W$ and the average concentration is less than the $DCGL_W$	Conduct the Sign test and the EMC test

6 COMPLIANCE WITH RADIOLOGICAL CRITERIA FOR LICENSE TERMINATION

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. Although an overview of the process for developing radiological release criteria is provided in this chapter, detailed information on example experiences at various sites is included in Appendix F.

It should be noted that this chapter contains values for certain input parameters (such as the depth to the groundwater table) that were used in the dose modeling computer codes by certain licensees. The reader needs to be aware that these types of values may be site-specific and may not be appropriate for your site.

6.1 U.S. NRC Site Release Regulations and Guidance

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

The NRC site release criteria (Reference 12) 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 meet the As Low As Reasonably Achievable (ALARA) criteria.

"Unrestricted use" means that there are no restrictions on the use of the site after the operating license is terminated. 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 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 17). Likewise, information on the RESRAD family of dose modeling codes is contained on the Argonne National Laboratory website (Reference 18). Both codes can be downloaded and used free of charge from their respective websites.

6.1.2 Evolution of Dose Model Scenarios

When the NRC made the LTR effective in 1996, there was limited guidance as to the dose models and scenarios that could be used. The only scenarios explained in the NRC guidance were based on unrestricted use of the site. The scenarios were also very conservative as described below.

6.1.2.1 Resident Farmer Scenario

This scenario assumes that a family resides at the site after license termination. This family grows all its food on the site including meat, fish and vegetables and utilizes a well drilled on site to obtain its drinking and irrigation water. To determine DCGLs, the NRC D&D code or the Residual Radioactivity (RESRAD) dose modeling computer code developed by Argonne National Laboratory were used. The D&D code can only calculate dose from soil while the RESRAD code can calculate doses for all pathways that are the result of residual radioactivity in soil and/or groundwater. Figure 6-1 shows an illustration of the various dose pathways for the Resident Farmer Scenario included in the RESRAD dose modeling code. Due to uncertainty in obtaining NRC approval of the model results, the very conservative, default input parameters were often used by utilities in the late 1990s and early 2000s. As will be discussed this approach resulted in very low DCGLs which in some cases required extensive remediation to achieve.

As information on the D&D code is readily available on the Oak Ridge Associated Universities website (Reference 17), detailed information on this code will not be provided in this report. Likewise, information on the RESRAD family of dose modeling codes is contained on the Argonne National Laboratory website (Reference 18).

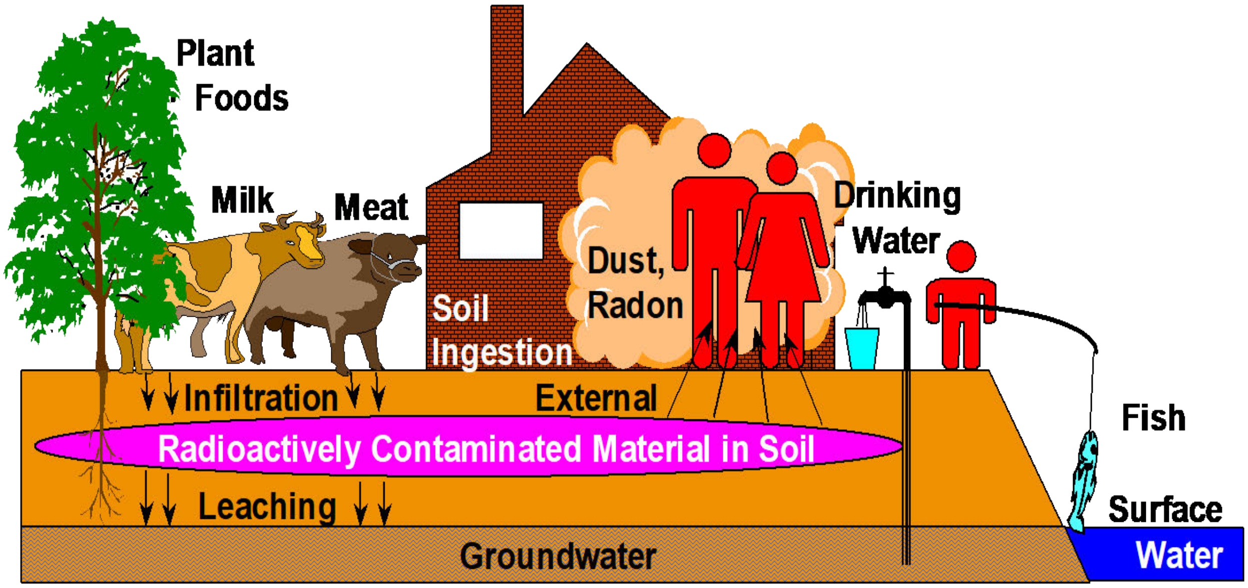


Figure 6--1 Dose Pathways of RESRAD Dose Modeling Code

6.1.2.2 Building Occupancy Scenario

In this scenario, workers in a concrete structure are assumed to be office workers that spend their 40+ hour work week in a room that was formerly part of the plant and has been released for use as an office. As with the dose from residual radioactivity in soil, the D&D and RESRAD-Build computer codes are commonly used to calculate these DGCLs. Building Occupancy DCGLs calculated using the D&D and RESRAD-Build (as used by Connecticut Yankee) code are compared in Table F-2 in Appendix F.

6.1.3 Revision to NRC Guidance on Dose Modeling

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 in SECY-03-0069, “Results of the License Termination Rule Analysis,” (Reference 19) 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.

This NRC Policy Issue, SECY-03-0069, 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 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.

In NRC Regulatory Issue Summary 2004-08, (Reference 10) the NRC Commissioners approved the recommendations of SECY 03-0069). 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 including Trojan, Connecticut Yankee, and Maine Yankee. Discussions on how these plants chose their land use scenarios are provided later in this chapter. Also discussed later is 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 SECY-03-0069. Discussion of the Big Rock Point experience is included later in this chapter.

6.1.4 NUREG 1757, "Consolidated Decommissioning Guidance"

The philosophies contained in SECY 03-0069 and RIS 2004-08 were incorporated into the 2006 revision of to NUREG 1757, "Consolidated Decommissioning Guidance," (Reference 9) and carried into revision 2 of the NUREG (Reference 14). 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:

- In most situations, there are numerous possible exposure scenarios in which future human exposure groups could interact with residual radioactivity. The compliance criteria in 10 CFR Part 20 for decommissioning does not require an investigation of all (or many) possible scenarios; its focus is on the dose to members of the critical group for the compliance exposure scenarios.

- The compliance exposure scenario is the exposure scenario that leads to the largest peak dose to the average member of the critical group from the mixture of radionuclides.

- 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.

- If the licensee bases its compliance exposure scenario on reasonably foreseeable land use scenarios which are not clearly bounding, the licensee should also identify less likely but plausible land use scenarios. These are scenarios that could lead to higher doses compared to the reasonably foreseeable land use scenario used to demonstrate compliance with the LTR criteria.

- The evaluation of less likely but plausible exposure scenarios ensures that, if land uses other than the reasonably foreseeable land use was to occur in the future, unacceptably high risks would not result.

NUREG-1757 defines the time frame for “reasonably foreseeable land uses” as within 100 years. Table 6-1 below from NUREG-1757 defines the different types of scenarios that are evaluated in determining site release limits (i.e., DCGLs). The appendices to NUREG-1757, include several scenarios that should be considered for analysis by a licensee. It should be noted that a licensee can present information to the NRC that one or more of these scenarios should be considered as implausible and therefore would not need to be analyzed.

Table 6-1
Description and Comparison of Dose Modeling Scenario Types

	Exposure Scenario Type	Description
Plausible Exposure Scenarios	*Compliance Exposure Scenarios (Results Compared to Dose Standards)*[2]	
	Screening	A predetermined exposure scenario that can be used with very high confidence, for most facilities, to demonstrate compliance with the radiological criteria for license termination without further analysis. It generally includes assumptions about land use or human behaviors that attempt to err on the side of higher doses. The screening exposure scenario for residual radioactivity on building surfaces is the building occupancy, and the screening exposure scenario for residual radioactivity in surface soils is the residential farmer.
	Bounding	An exposure scenario with a calculated dose that bounds the doses from other likely exposure scenarios. The building occupancy and residential farmer screening exposure scenarios would represent bounding exposure scenarios for most site-specific analyses for residual radioactivity on building surfaces and in surface or subsurface soils, respectively.
	Reasonably Foreseeable	Land use exposure scenarios that are likely within the next 100 years, considering current area land-use plans and trends. These exposure scenarios are site-specific.
	Other Exposure Scenarios (Results Used to Inform Decisions)	
	Less Likely but Plausible	Land use exposure scenarios that are possible, based on historical uses or trends, but are *not* likely within the next 100 years, considering current area land use plans and trends. These exposure scenarios are usually site-specific.
Implausible Exposure Scenarios	*Implausible Exposure Scenarios (No Analysis is Required)*	
	Implausible	Land uses that, because of physical or other compelling limitations, could not occur (e.g., residential land use for an underwater plot of land).

2 Any or all of the compliance scenarios can be used to demonstrate compliance with the radiological criteria for license termination. In general, greater support is needed to demonstrate compliance when using reasonably foreseeable exposure scenarios that have limited pathways, consumption rates, or occupancy times compared to the screening scenarios .

6.1.5 Realistic Dose Modeling Scenarios

6.1.5.1 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 6-2 (from NUREG-1757, Reference 9). depicts most of the typical exposure pathways for the Industrial Worker Scenario.

Figure 6--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

6.1.6 Site Future Use Decision Case Studies

6.1.6.1 Connecticut Yankee

The Haddam Neck Plant, (commonly called Connecticut Yankee or CY) was located on a 525-acre site in Haddam, Connecticut and housed a Westinghouse 4-loop PWR rated at 1,825 MWth and 619 MWe. This facility was operated by the Connecticut Yankee Atomic Power Company. CY began commercial operation in January 1968 and was permanently shut down in December 1996.

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. For the land area of the plant, Yankee Rowe also utilized the Resident Farmer Scenario as its dose modeling scenario for the same reasons as CY.

6.1.6.2 Big Rock Point - Modified Resident Farmer Scenario

Big Rock Point was a 75 MWe Boiling Water Reactor (BWR) located on a 564-acre 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 20).

- The critical group for site-specific analysis of the Industrial Area at Big Rock Point was a modified resident farmer who moved onto the site.
- Some of this resident farmer's diet consists 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 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 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.

6.1.6.3 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 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 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 in addition to the natural gas fired power plant on the site. These site use decisions allowed the Industrial Worker Scenario to be used in determining the release limits for the Rancho Seco site.

6.1.6.4 Zion Resident Farmer Scenario (Reference 15)

For the Zion site the soil DCGLs were calculated using the standard RESRAD parameter selection and uncertainty analysis methods as defined in NUREG 1757 (Reference 9). To account for dose from shallow and deep contaminant, two different DCGLs were calculated for surface soil ($DCGL_{SS}$) and subsurface soil ($DCGL_{SB}$).

Surface soil weas defined as that contained in a 0.15 m depth from the surface. Subsurface soil was defined as that contained in a 1 m depth from the surface. These definitions apply to a continuous soil column for the surface downward. There was no expectation of subsurface contamination in the geometry of a clean soil layer over a contaminated soil layer at depth. The subsurface soil DCGLs could be conservatively applied to any soil depth greater than 0.15 and less than 1 m.

6.1.6.5 LaCrosse (Reference 21)

As the LaCrosse site contained a thermal power plant that would continue to operate, the site used the Industrial Worker Scenario to determine the DCGLs for soil. Unlike the methodology used for Zion, LaCrosse developed only one soil DCGL for soil to a depth of 1 m. This was done for conservatism and to optimize remediation efficiency by reducing the potential for delays or unnecessary remediation if contamination with a thickness somewhat greater than 0.15 m was encountered.

As with the Industrial Worker Scenario described for Rancho Seco above, the main parameter affecting dose compared to the Resident Farmer Scenario is the occupancy time. Consistent with NUREG-6697 (Reference 22) and considering a 5-day work week, 50 weeks per year, yielded 2190 hours per year on site. Per the RESRAD user's manual, 75% of the work time was assumed to be indoors and 25% outdoors.

Drinking water intake rates from NUREG-5512 (Reference 23) for a residential user were adjusted to the 250 days per year that the industrial worker was assumed to be on site. NUREG 5512 was also used as a source for the inhalation rate parameter for light industry of 1.4 m3/hr. This was applied to an on-site time of 2190 hrs/yr to yield a parameter value of 3066 m3/yr.

As described in the NRC guidance contained ion NUREG-1757, in addition to the "reasonably foreseeable" scenario (i.e., industrial worker), LaCrosse analyzed two alternate land use scenarios as follows:

6.1.6.5.1 Recreational Land Use Scenario

A qualitative evaluation of a recreational land use scenario concluded that the dose would be less than that calculated for the industrial use scenario because occupancy time would be less than that assigned to an industrial worker. In addition, if a water supply well were to be installed in the recreational land use scenario, the recreational user's intake rate from the well would be less than assumed for the industrial worker.

6.1.6.5.2 Resident Gardner Scenario

The second alternate use scenario evaluated was the residential gardener. This scenario was considered to be a "less likely but plausible" scenario as described in Table 6-1 above. RESRAD weas used, in conjunction with an Excel spreadsheet, to calculate the dose for the Resident Gardener alternate scenario. NUREG 1757 states that if the peak dose from a less likely but plausible scenario is "significant," then greater assurance that the scenario is unlikely would be necessary.

LaCrosse evaluated the Resident Gardener dose after 30 years decay for soil areas and basements left on site. The maximum soil dose was 27.07 mrem/yr. The maximum basement doses were calculated as 28.4 mrem/yr and 34.9 mrem/yr for the Reactor Building and the Waste Gas Tank Vault, respectively. These doses are not considered significant and therefore greater assurance that these scenarios will not occur was not necessary. LaCrosse adjusted the soil and basement DCGLs used for FSS to ensure that no alternate scenario dose could exceed 25mrem/yr.

6.2 Dose Modeling to Determine Site Release Limits

This section will discuss the general basis used at decommissioning nuclear power plants in the United States for selecting dose modeling scenarios for land areas. These scenarios are then used to develop radionuclide concentration site release limits that meet dose-based site release criteria. Appendix F contains some examples of dose modeling experiences at nuclear power plant sites since the implementation of the License Termination Rule in 1996.

6.2.1 Land Areas

Once the decision on the future use of the site has been made the corresponding scenario is used to determine the radionuclide concentration limits that meet the site release criteria. 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 of options in determining the site release limits to be used during different phases of the decommissioning planning and implementation. While Appendix F describes options that require dose modeling, the following describes options that determine site release limits for land areas that do not require dose modeling.

6.2.1.1 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, Table H.2 (Reference 9). The soil screening values for most of the radionuclides that have been found to be significant during past decommissioning projects are listed in this NUREG. These screening values were determined using the NRC D&D dose modeling code, which was developed by the Sandia National Laboratories (information on the D&D code is contained in Reference 17). The default parameters of the D&D code are based on the Resident Farmer Scenario 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 NUREG 1757, 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 effect 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. If 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 – several 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. Of 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 been subtracted from the 10 mrem/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 to obtain the adjusted values.
- 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 several survey units. Each survey unit must meet the DCGLs approved for the site. The NRC also noted in NUREG-1757 (Reference 9) 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. 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.

6.2.1.2 Adjusting NRC Screening Values for Potentially Contaminated Groundwater

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

- [redacted]
- [redacted]

[redacted]

- [redacted]
- [redacted]
- [redacted]

Considering that the most likely detected radionuclides at a power plant are H-3 and Sr-90, a conservative bounding approach for setting site release limits for soil is the following:

- As the TEDE dose at the MCL concentrations for both H-3 and Sr-90 is approximately 0.8 mrem/yr (and the “sum of fractions” rule applies to showing compliance with the MCLs), reduce the NRC soil screening values to reflect a projected dose of 2 mrem/year from groundwater contamination at the time of decommissioning. This additional 1.2 mrem/year of allotted dose for radionuclides other than H-3 and Sr-90 would be expected to cover practically all situations that would occur at a power plant site.

- An additional reduction of 2 mrem/year in the dose allotted to soil (for a total reduction of 4 mrem/year) should be made to account for the potential of radionuclides leaching out of concrete basements as was required during the Connecticut Yankee, Maine Yankee and Yankee Rowe decommissioning (the site with the highest dose from concrete basements was CY at 1.58 mrem/year).

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 could be reevaluated. The approach of adjusting the NRC screening DCGL is consistent with the guidance in NUREG-1757 (Reference 9) which states in the footnote on the bottom of page I-5:

> *If surface water or groundwater are contaminated, it may still be possible to use screening values if the dose contributions from the residual radioactivity in the surface water or groundwater are separately considered.*

6.2.2 Building Surfaces

As discussed for land areas in the previous section, once the decision on the future use of the site has been made the corresponding scenario is used to determine the radionuclide concentration limits that meet the site release criteria. 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.

Like the case for land areas, a plant site has a number of options in determining the site release limits to be used during different phases of the decommissioning planning and implementation. Appendix F describes options that require dose modeling and the use of NRC Screening Values for building surfaces that do not require dose modeling.

7 UPDATE ON SITE-SPECIFIC DECOMMISSIONING COSTS

10 CFR 50.82(a)(9)(ii)(F) requires a licensee to provide an updated site-specific decommissioning cost estimate (DCE) that includes an estimate of the cost of remaining decommissioning work as part of its LTP. This update must reflect any changes that occurred since the original DCE was submitted. The update should also include the effects of inflation, and changes in radioactive waste disposal costs. If little decommissioning has been completed, and inflation and disposal costs have not changed, the cost estimate originally submitted pursuant to 10 CFR 50.82(a)(4)(i) and 10 CFR 50.82(a)(8)(iii) may be acceptable.

The update should reflect the status of the facility at the time of LTP submittal and the licensee's plans for how remaining decommissioning activities will be completed. In accordance with 10 CFR 50.75, the nuclear decommissioning trust (NDT) must be funded to the amount of the cost estimate, and the updated site-specific cost estimate must address the remaining activities necessary to complete decommissioning to ensure that sufficient funds are available.

Therefore, the LTP should include an estimate of the decommissioning costs remaining at the time of LTP submittal, and a comparison of the estimated remaining costs with the present funds set aside for decommissioning. If there is a deficit in present funding, then indicate the means for ensuring adequate funds to complete the decommissioning. Regulatory Guide 1.159, "Assuring the Availability of Funds for Decommissioning Nuclear Reactors," (Reference 25) provides detailed guidance on methods for estimating decommissioning costs and on financial assurance mechanisms that are acceptable to the NRC staff. If the LTP indicates that the licensee will provide assurance of funding by a surety method, insurance, or other guarantee, then the financial assurance instrument should remain in effect until the NRC has terminated the license.

As a minimum, the decommissioning cost estimate should evaluate the following seven cost elements:

(1) cost assumptions used, including a contingency factor

(2) major decommissioning activities and tasks

(3) unit cost factors

(4) estimated costs of decontamination and removal of equipment and structures

(5) estimated costs of waste disposal, including applicable disposal site surcharges and transportation costs

(6) estimated final survey costs

(7) estimated total costs

The cost estimate should focus on the remaining work and provide details for each activity associated with the decommissioning, including the costs of labor, materials, equipment, energy, and services. The cost estimates should be based on credible engineering assumptions that are related to all remaining major decommissioning activities and tasks. RG 1.159 provides detailed guidance on methods that NRC finds acceptable for estimating decommissioning costs.

The cost estimate should include the cost of the planned remediation actions, the cost of transportation and disposal of the waste generated by the actions, and other costs that are appropriate for the planned actions. NUREG-1307, “Report on Waste Burial Charges: Changes in Decommissioning Waste Disposal Costs at Low-Level Waste Burial Facilities,” (Reference 26) issued January 2013, provides information on estimating waste disposal costs. The cost estimate should not include any credit for the salvage value of equipment.

The introduction to chapter 7 of the LTP should provide a discussion of the original DCE and the basis used to develop the updated estimate. Cost information may be company proprietary and should be protected as such by submitting a proprietary version of chapter 7 along with a redacted version.

7.1 Decommissioning Cost Estimate

In this section, provide the updated site-specific DCE with a breakdown of the remaining costs to complete the decommissioning process and to release all portions of the site for unrestricted use, except for the area required for the ISFSI and associated protected area.

The following subsections of the LTP will present a description of how the cost estimate was prepared and a summary and breakdown of the estimated costs.

7.1.1 Cost Estimate Description and Methodology

The cost model includes elements for estimating distributed and undistributed costs. Distributed costs are activity specific and include planning and preparation costs as well as costs for decontamination, packaging, disposal, and removal of major components and systems. For example, costs for the segmentation, packaging, and disposal of the reactor internals are distributed costs. Undistributed costs, sometimes referred to as collateral costs, are typically time dependent costs such as utility (licensee) and decommissioning general contractor (DGC) staff, property taxes, insurance, regulatory fees and permits, energy costs, and security staff.

The methodology for preparing cost estimates for a selected decommissioning alternative requires development of a site-specific detailed work activity sequence based upon the plant inventory. The activity sequence is used to define the labor, material, equipment, energy resources, and duration required for each activity. In the case of major components, individual work sequence activity analyses are performed based on the physical and radiological characteristics of the component, and the packaging, transportation, and disposal options available.

In the case of structures and small components and equipment such as piping, pumps, and tanks, the work durations and costs are calculated based on UCFs. UCFs are economic parameters developed to express costs per unit of work output, piece of equipment, or time. They are developed using decommissioning experience, information on the latest technology applicable to decommissioning, and engineering judgment.

7.1.2 Summary of the Site-Specific Decommissioning Cost Estimate

The cost estimate includes the costs for license termination costs (corresponding to 10 CFR 50.75(c) requirements), SFM costs (corresponding to 10 CFR 50.54(bb) requirements), and site restoration costs (corresponding to activities such as clean building demolition and site grading etc.). A breakout of the cost for each part of the decommissioning program is provided in a table.

7.1.3 License Termination Costs

Consistent with the NRC definition of decommissioning under 10 CFR 50.75(c), the decommissioning costs under this category consider only those costs associated with normal decommissioning activities necessary for release of the site (other than the ISFSI) for unrestricted use. It does not include costs associated with the disposal of non-radiological materials or structures beyond those necessary to terminate the Part 50 license or the costs associated with construction or operation of an ISFSI.

The remaining decommissioning scope of work included in this estimate is described in detail in other chapters of this LTP. Overall, that work scope includes completion of the removal, transportation, and disposal of the major components; completion of the removal, transportation, and disposal of the remaining equipment; decontamination and/or bulk demolition of radiological impacted structures and transportation and disposal of the resulting radioactive wastes; performance of the FSS and associated license termination activities. The estimated costs include the labor, equipment, materials, services, and fees needed to conduct the work. The estimated cost also includes all the program support activities and services necessary to manage and safely carry out a large-scale dismantlement and demolition project. These program support activities include project management, work controls and site administration; technical support services, such as radiation protection, safety, engineering, security, QA/QC, environmental monitoring, waste management and decommissioning subject matter experts needed to support the project.

A high-level breakdown of the estimated cost by phase is provided in a table.

7.1.4 Spent Fuel Management Costs

The costs to construct and operate an ISFSI and other spent fuel management costs are not considered by the NRC staff as part of decommissioning costs. Nevertheless, as there is significant interest by many stakeholders in these costs, they should be described in this section.

7.1.5 Site Restoration Costs

The estimated cost for the anticipated site restoration work scope is discussed. Overall, that work scope includes clean building demolition, backfilling of any open excavations or void spaces, non-rad environmental remediation, and final grading and stabilization against erosion.

7.1.6 Contingency

Contingencies are applied to cost estimates primarily to allow for unknown or unplanned occurrences during the actual program (e.g., increased radioactive waste materials volumes over that expected, equipment breakdowns, weather delays, and labor strikes). This is consistent with the definition provided in the DOE Cost Estimating Guide (Reference 27). Contingency "covers costs that may result from incomplete design, unforeseen and unpredictable conditions, or uncertainties within the defined project scope. The amount of contingency will depend on the status of design, procurement, construction, and the complexity and uncertainties of the component parts of the project. Contingency is not to be used to avoid making an accurate assessment of expected costs."

7.2 Decommissioning Funding Plan

Include a discussion of the existing decommissioning trust fund status using the same values provided in the annual decommissioning funding assurance report required by 10 CFR 50.75 and 10 CFR 50.82(a)(8)(v).

8 SUPPLEMENT TO THE ENVIRONMENTAL REPORT

8.1 Introduction

This chapter of the LTP should address each of the criteria from 10 CFR 50.82(a)(9) and 10 CFR 50.82(a)(10), and the related radiological criteria from Subpart E of 10 CFR Part 20 for unrestricted or restricted release of the site. One of these criteria is to provide a supplement to the environmental report, in accordance with 10 CFR 51.53, "Postconstruction Environmental Reports," that describes any new information or significant environmental change associated with the licensee's proposed termination activities.

Regulatory Guide 1.170, Rev. 2, "Standard Format and Content of License Termination Plans," states that pursuant to 10 CFR 50.82(a)(9)(ii)(G), the licensee should submit a supplement to the environmental report describing any new information or significant environmental change associated with the site-specific termination activities. The supplement to the environmental report should do the following:

a. Describe in detail the environmental impact of the site-specific termination activity.

b. Compare the impact with previously analyzed termination activities (see NUREG-0586, "Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities," Supplement 1; "Regarding the Decommissioning of Nuclear Power Reactors," issued November 2002); and

c. Analyze the environmental impact of the site-specific activity. Include alternative actions and any mitigating actions.

There are two points during the decommissioning process when the licensee performs an evaluation of environmental impacts. The first evaluation occurs when the licensee must submit a PSDAR to the NRC (within two years following permanent cessation of operation). The PSDAR must include a discussion that provides the reasons for concluding that the environmental impacts associated with the licensee's planned site-specific decommissioning activities will be bounded by an appropriate previously issued environmental assessment, including the NUREG-0586 Supplement. If the licensee identifies environmental impacts that are not bounded by a previous NRC environmental assessment, the licensee must address the impacts in a request for a license amendment regarding the activities. The licensee must also submit a supplement to its environmental report (ER) that describes and evaluates the additional impacts. The NRC reviews the supplement to the ER in conjunction with its review of the license-amendment request.

The second evaluation is near the end of decommissioning at the time when the licensee applies for license termination. In accordance with 10 CFR 50.82(a)(9), a licensee must submit its LTP at least 2 years before the anticipated termination date of the license. The LTP must be a supplement to the Final Safety Analysis Report or its equivalent for the facility and is submitted as a license amendment. The NRC requires an environmental review as part of the review of the license amendment request. Thus, the LTP must include a supplement to the ER that describes any new information or significant environmental change associated with the licensee's proposed termination activities. The NRC staff will also rely upon this supplement as a basis for determining if anticipated decommissioning impacts require an additional review.

8.2 General Guidance

This LTP section is written following the outline in NUREG-0586, Supplement 1, Volume 1, published in November 2002. Each section has four parts:

a. Regulations -Identifies statutes, regulations, or limits relevant to the issue.

b. Potential impacts from decommissioning activities -Discusses possible impacts related to the issue and defines, where appropriate, the terms detectable and destabilizing for the issue.

c. Evaluation -Describes analysis and professional judgement used to estimate whether an activity or group of activities is likely to make a noticeable impact on the environment, considering the available data. If an impact is likely, mitigation measures that can be taken to avoid the impact are evaluated. If an impact cannot be avoided, a determination is made as to whether the impact is likely to destabilize the resource.

d. Conclusion - Provides the conclusion on significance (SMALL, MODERATE, LARGE) and applicability (generic or site-specific) of impacts to the issue.

8.3 Lessons Learned

[redacted]

8.4 Land Use - Offsite Land Use Activities

Land use outside of the plant disturbed area needs to be discussed if either on or off the licensed footprint. This would include road and rail modifications, storage locations, transfer stations or the use of borrow pits.

8.5 Aquatic Ecology – Offsite Effects Beyond the Operational Area

Groundwater use that affects an offsite aquifer needs to be discussed to ensure the impacts do not adversely affect neighboring use of the same aquifer. Temporary surface water diversions or coffer dams may adversely affect downstream fish ladders or aquatic habitat. Also, disturbance of riverbanks and regarding the site for stormwater runoff can alter the aquatic habitat. Special care is needed to ensure there are no adverse impacts.

8.6 Terrestrial Ecology

Terrestrial habitats can be affected by decommissioning such as peregrine falcon, osprey or seagull nests on plant structures, communication poles and towers. Good communication with state and federal environmental agencies is necessary to relocate or remove the bird nests during decommissioning. It is also important to relocate other animals such as turtle eggs, snake dens, etc. when site preparations for decommissioning uncover their habitat. A potential destabilizing effect can be turned into a positive environmental impact for the project when done properly.

8.7 Threatened and Endangered Species

The current list of threatened and endangered species is required for the supplement to the environmental report. The list is updated on a continual basis by state environmental agencies so relying on an older list is not sufficient. Special care must be taken to minimize any impacts to these species during decommissioning by avoiding their habitat near the plant. If adversely affecting the habitat is unavoidable, the species may need to be relocated. Decommissioning work may be restricted during certain times of the year if decommissioning activities could adversely affect offspring. Close communication with state and federal environmental agencies is critical so as not to delay the project.

8.8 Environmental Justice

Decommissioning may affect transportation routes, transfer station locations and site organizations. An analysis is required to determine if there are any adverse impacts to minority or low-income populations in the vicinity of the decommissioning project. Use of local labor and local business is recommended for positions that can be filled by the local community. Reference 28 provides the NEI policy statement on environmental justice.

8.9 Cultural and Historic Activities Beyond the Operational Area

The State Historic Preservation Office (SHPO) maintains a list of cultural and historic locations on the site or in the vicinity. If such locations are identified, field surveys and evaluations by archeologists and historians may be required. It is important to identify these locations at the start of the project to avoid the potential of inadvertently damaging or destroying these locations during decommissioning. It is also important to begin the required field surveys, investigations, and potential remediation prior to or by the time of LTP submittal to avoid any potential delays in LTP approval by the NRC. Frequent communication with the SHPO are important to ensure all stakeholders agree as to the actions taken.

9 FINAL STATUS SURVEY REPORTING

9.1 Introduction

The implementation of the FSS to achieve the defined end state is one of the most critical phases of the decommissioning process. As described in section 5 of this document, the general (high-level) acceptance criteria for the FSS Plan are described in Section 2.5.1 of NUREG-1700. This NUREG references another NRC guidance document (NUREG-1757, Vol. 2) for more detailed acceptance criteria. Appendix A of NUREG-1700 also provides a general description of the Acceptance Review Checklist for an FSS Report. NUREG-1757, Section 4.5, provides a much more detailed description of "Areas of Review" for the FSS Report. Regulatory requirement (CFR) documents are cited as well as the MARSSIM (NUREG-1575). The intent of this section is to provide some practical observations that may be useful in simplifying the FSS Report review and RAI process for both NRC and licensees.

As stated in NUREG-1757, Volume 2, the acceptance criteria for an FSS report are based on the following federal requirements: 10 CFR 20.1402, 20.1403, 20.1501, 30.36(j)(2), 40.42(j)(2), 70.38(j)(2), and 72.54(i)(2). The primary regulatory guidance is NUREG–1575, "Multi-Agency Radiological Survey and Site Investigation Manual" (MARSSIM).

The importance of FSS documentation cannot be underestimated. This documentation ultimately serves as the data and information that NRC will utilize to develop their Safety Evaluation Report (SER) for license termination.

Chapter 5 of the MARSSIM emphasizes the importance of technically defensible documentation and states: "Documentation of the FSS should provide a complete and unambiguous record of the radiological status of the survey unit relative to the established DCGLs. In addition, sufficient data and information should be provided to enable an independent re-creation and evaluation at some future time. Much of the information in the FSS report will be available from other site remediation documents; however, to the extent practicable, this report should be a stand-alone document with minimum information incorporated by reference."

[redacted] Guidance is provided on FSS reports in the following documents:

- NUREG-1575, Revision 1, "Multi-Agency Radiation Survey and Site Investigation Manual" (MARSSIM)
- NUREG-1757, Volume 2, Revision 2, "Consolidated Decommissioning Guidance - Characterization, Survey, and Determination of Radiological Criteria - Draft Report for Comment"

[redacted]

Note that the FSS Plan (chapter 5 of the LTP) serves as the roadmap for FSS at a given site. Therefore, the FSS report content requirements should be clearly delineated in the LTP.

One notable exception to guidance (which is not currently addressed in NRC guidance) is the need for early (“at-risk”) FSS. This pertains to FSS that must be performed prior to submittal and/or approval of the LTP, such as excavation/backfill of areas necessary in order to prevent major delays in project D&D and remediation work. NEI strongly recommends that licensees request, and NRC provides the necessary support from the Region, HQ PMs, and HQ technical reviewers in order to allow for such work.

9.2 Final Status Report Content

FSS results (including individual survey unit packages and FSS reports) are ultimately based on the DQO process described in the FSS Plan (chapter 5 of the LTP). The licensee’s FSS Plan should clearly define the content of survey unit and FSS reports.

NUREG-1757, Section 4.5, lists items that require Minimum Technical Review, and the criteria for NRC selecting survey units for Detailed Technical Review. The licensee should review this information in order to be prepared for possible submittal requests.

NUREG-1757, Section 4.5, states that the following minimum information should be included in an FSS report.

- An overview of the results of the FSS
- A summary of the DCGLs for the facility (if DCGLs are used)
- A discussion of any changes that were made in the FSS from what was proposed in the DP *(or LTP)* or other prior submittals
- A description of the method by which the number of samples was determined for each survey unit
- A summary of the values used to determine the number of samples and a justification for these values
- The survey results for each survey unit including the following:
    - the number of samples taken for the survey unit
    - a description of the survey unit, including (a) a map or drawing of the survey unit showing the reference system and random start systematic sample locations for Class 1 and 2 survey units, and random locations shown for Class 3 survey units and reference areas; (b) discussion of remedial actions and unique features, and (c) areas scanned for Class 2 and 3 survey units
    - the measured sample concentrations, in units that are comparable to the DCGLs
    - the statistical evaluation of the measured concentrations

  - judgmental and miscellaneous sample data sets reported separately from those samples collected for performing the statistical evaluation
  - a discussion of anomalous data including any areas of elevated direct radiation detected during scanning that exceeded the investigation level or any measurement locations in excess of DCGLW; and
  - a statement that a given survey unit satisfied the DCGLW and the elevated measurement comparison if any sample points exceeded the DCGLW

- A description of any changes in initial survey unit assumptions relative to the extent of residual radioactivity (e.g., material not accounted for during site characterization)
- A general description of how the survey results achieve the ALARA requirements of Reference 12.

Other information recommended to be included in survey unit and FSS reports include:

- survey results for any in-process surveys performed to support FSS (including surveys performed as a result of elevated activity, remediation activities, etc.)
- QC results (for split and duplicate samples, instrumentation trending charts, etc.)
- any changes from the original FSS survey design
- Data Quality Assessment (DQA) conclusions
- any anomalies encountered during performance of the survey or in the sample results
- Summary of conclusion as to whether the SU satisfied the release criteria

The above information is typically included in individual survey unit reports (sometimes referred to as release records). A compilation of individual survey unit reports is then presented as an FSS report. The reports may be presented to NRC in phases (as the work is completed) to 1) allow for early feedback (via RAIs) from NRC on the format and technical content on the FSS data and reports, and 2) streamline and expedite NRC reviews.

Per NUREG-1757, **the FSS is adequate if it meets the criteria in the following**:

- MARSSIM Section 5.5.2 for the acceptable number of samples
- Appendix D, NUREG-1757 for information on survey data quality and reporting
- Section A.10 from Appendix A of NUREG-1757 for information on determining compliance, and
- MARSSIM Sections 8.3, 8.4, and 8.5 for interpretations of sample results

Conciseness and cohesiveness in survey unit and FSS reports are critical to allow for ease of NRC reviews. It is recommended that supporting documentation is compiled and filed by the licensee and provided for NRC reviews as requested rather than being included in the survey unit and FSS reports. Ideally any quality questions/issues are evaluated by NRC in-process (either via information audits, procedure reviews, or in-process inspections) to avoid extensive questions and simplify the RAI process following the submittal of the FSS report(s). Examples of items recommended to be reviewed by NRC in-process include:

- The licensee's QA Plan, including quality control and assurance requirements for onsite field and laboratory instrumentation, and offsite laboratories
- Evaluation of subsurface survey/sampling requirements/techniques
- Question related to surveys/sampling/process for use of onsite media (soil, crushed concrete, etc.) for fill material
- Licensee process for demonstration of meeting compliance dose and ALARA requirements
- Application of alternate dose scenarios for small source terms (e.g., DRPs) that are not uniformly distributed and therefore not included in compliance dose calculations
- NRC expectations on documentation that should be submitted for formal review (as approved in the LTP)

To achieve this level of FSS report standardization, **early and often communications** with both the NRC region and headquarters staff are critical.

In addition, early engagement with the NRC's confirmatory survey contractor will contribute to identification of potential procedure and implementation issues early in the project and avoid delays in back-end reviews of FSS documentation. Additional discussion of confirmatory surveys is provided in section 9.3 below.

Note that the final dose compliance calculation (which demonstrates compliance with the site-specific dose criterion to a member of the public) is typically provided as part of a separate report for request for partial site release (see section 1.1.5).

9.3 Role of NRC Independent Oversight and Confirmatory Measurements

The LTP specifies the FSS process including the licensee quality assurance plan for both the process and the resultant data. As a license condition, specified LTP processes and controls are implemented through licensee approved procedures for both implementation and licensee quality oversight. As with operating nuclear power plants, the NRC oversight role is to assure the licensee is adhering to the LTP and approved procedures. This oversight along with independent confirmatory measurements performed by or for the NRC during applicable phases of decommissioning should allow for minimal backend report documentation of processes and excessive raw data. Refer to section 9.2 for recommended FSS Report content.

9.3.1 NRC Oversight

NRC Inspection Procedure 83801 lists objectives for inspection of remedial and final surveys at permanently shut down reactors. Table A of the procedures provides a description of the inspection focus for each licensee activity (including Remedial Action Support Surveys, FSS In Progress, FSS Completed, and FSS Reports). The table describes the inspection purpose and process based on guidance from NUREG-1757. Appendix A of the procedure is a Final Status Survey Program Inspection Checklist that could serve as a valuable tool for the Regional Inspector(s) and the NMSS Technical Reviewers to perform in-process reviews of the licensee's FSS program during the procedure development and implementation phase. As described in several areas of this document licensees request in-process engagement (during active D&D and FSS) of the Regional Inspectors and NMSS Project Manager, along with the Technical Reviewers who will ultimately review the FSS Reports and generate the Safety Evaluation Report (SER). This recommendation would reduce or eliminate the necessity for licensees to provide the same or similar information that was previously provided following the submittal of the FSS reports (during the RAI phase); as the FSS Report technical reviewers will be engaged and can raise issues/concerns prior to or during the implementation phase of FSS.

9.3.2 Confirmatory Surveys

A Confirmatory Survey, as described in the glossary in NUREG-1757, is a survey conducted by the NRC or its contractor to verify the results of the licensee's FSS. Confirmatory surveys are invaluable to provide confidence that the licensee's data meets the specified residual radioactivity levels for release (i.e., DCGLs). The licensees have observed that there are some areas for gains in efficiency in the confirmatory survey process.

- Confirmatory survey design should follow that of the approved licensee FSS design (scan speed, sample density, etc.)
- Confirmatory survey data/results should be provided to the licensee in a timely manner in order for the licensee to take appropriate action.
- In-Process Surveys performed either during or prior to completing remediation activities (e.g., for excavations/backfill performed prior to submittal and/or approval of the LTP), or during the licensee's performance of the FSS (as described in NRC Inspection Procedure 83801) greatly streamline the required review process for FSS reports given that issues can be identified and corrected prior to the issuance of such reports.

9.3.3 Optimizing the Role of NRC PM and Tech Reviewers

Section 4.5.1.2 of NUREG-1757 states the following: "In addition, the NRC reviewer may need to obtain previous NRC-generated reports regarding the FSS, including but not limited to inspections, confirmatory surveys, and SERs for any safety evaluation reports that may have addressed the FSS plan."
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APPENDIX A. APPLICATION OF ADVANCED TECHNOLOGIES TO SHOW COMPLIANCE

Although most plants have conducted FSS using a combination of fixed measurements and scans, MARSSIM also allows for use of advanced survey technologies if these techniques meet the applicable requirements for data quality and quantity. Advanced survey techniques may be used alone or in combination with fixed measurements and scans to assess a survey unit. For Class 1 and Class 2 units, two conditions must be met for advanced technologies to be employed as the only survey technique:

- An acceptable fraction of the survey unit surface area must be analyzed by the instrument.
- The minimum detectable concentration (MDC) for the measurements must be an acceptable fraction of the site release (no more than 50%).

In cases where these coverage requirements cannot be achieved by advanced survey technology or where the MDC is too large relative to the applicable limits, the survey can be augmented with fixed measurements and traditional scans. Advanced technologies may be used for judgmental assessments in Class 3 areas if the MDC is less than the site release limits.

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A.2.1 Rancho Seco Containment Dome Survey

One of the more successful uses of in situ gamma spectroscopy (in-situ) is the FSS of the inside of the containment dome at Rancho Seco. The interior surface of the Rancho Seco reactor building dome consists of the painted steel liner that extends from the spring line, located approximately 6 m above the refueling floor, to the top of the dome which is approximately 19 m above the refueling floor. The inside of the dome has a total surface area of approximately 1941 m^2. The liner had become contaminated during plant operation and had been subsequently decontaminated. However, the liner surface was not directly accessible to personnel for performing surveys to determine the effectiveness of the decontamination efforts.

Previous experience in performing the FSS of the inside of the dome at the Trojan plant involved using conventional scanning performed from a large scaffolding structure that had been constructed on top of the polar crane. Challenges with surveying such structures along with low site release limits made several decommissioning plants in the U.S. decide to dispose of all the above grade portions of the containment dome as radioactive waste.

The Rancho Seco approach allowed the survey of the liner to be performed without a loss of sensitivity and without having to place personnel at risk to build scaffolding on the crane, climb the scaffolding to perform surveys, conduct remediation, if necessary, then disassemble the scaffolding once the survey was complete. By employing in situ gamma spectroscopy with a wireless-configured Multi Channel Analyzer (MCA), a small platform was placed on top of the polar crane which supported a remote-controlled man lift that could position the in-situ detector at the necessary locations for performing the surveys of the dome interior surface without a technician riding the manlift.

The survey design called for overlapping 28 m^2 fields of view with a source-to-detector distance of 3 m. The survey was performed using a characterized, 40% relative efficiency HPGe detector.

The count times were set (typically 10 minutes) to achieve an MDC of approximately 2200 dpm/100 cm^2 (0.37 Bq/cm^2).

MARSSIM (Section 5.5.3.2) states that in situ gamma spectroscopy may be used where gamma emitting radionuclides are present to demonstrate compliance with the release criterion. It also states in MARSSIM Section 5.3.5 that “if the equipment and methodology used for scanning is capable of providing the same quality as direct measurements (e.g., detection limit, location of measurements, ability to record and document results), then scanning may be used in place of direct measurements.” Rancho Seco was able to make this demonstration of sufficient instrument sensitivity to the satisfaction of the NRC.

If the survey had been designed in the standard manner using conventional beta sensitive meters, 272 direct gross beta measurements would have been required, each covering a total area of 2.72 m^2 and the entire surface of the dome scanned with handheld meters. The in-situ survey performed consisted of 110 individual measurements covering 3113 m^2. Scan coverage was 160% of the total surface area. Rancho Seco concluded the survey results met the requirements for a Class 1 survey and demonstrated that the residual activity on the upper portion of the containment liner met the site release criterion.

In-situ gamma spectroscopy using portable germanium detectors was shown to be a very effective instrument for the radiological survey of concrete structures during the decommissioning of the Rancho Seco plant. Its use greatly increased the safety of the survey personnel while reducing the overall survey schedule. The instruments were also shown to be sufficiently sensitive to meet the design requirements of FSS. In addition to the survey of the liner, Rancho Seco also used in-situ gamma spectroscopy to survey the containment basement floor and several smaller rooms.

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A.3.1 Final Status Survey of Plant Effluent Water Course: Class 2, Survey Unit # 1

The area called the Plant Effluent Water Course discussed in the last section was the release point for liquid effluents from the plant. Figure A-1 shows the location of this area in relation to the plant industrial area. The area was impacted by multiple planned and unplanned liquid releases. Figure A-2 shows the locations of in-situ gamma spectroscopy counts that were conducted in this area.

Figure A-1
Arial View of Rancho Seco Plant Highlighting Plant Effluent Water Course Location

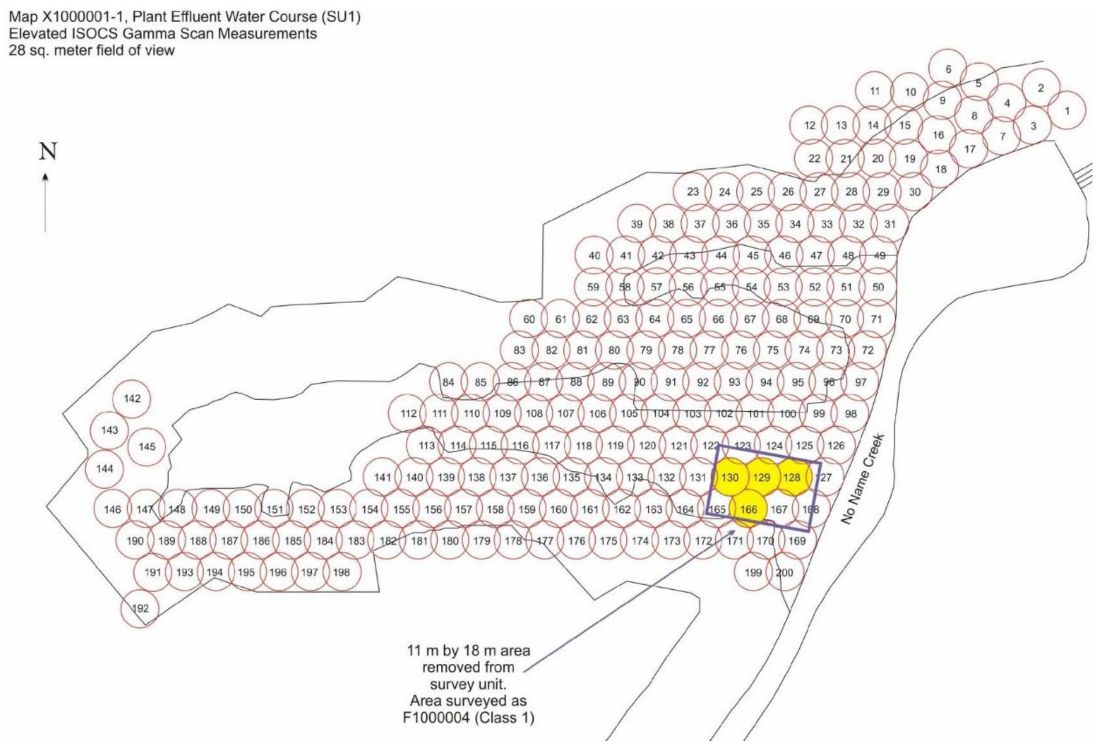

Figure A-2
Locations of In-situ Gamma Spectroscopy Counts at Rancho Seco Plant Effluent Water Course

A.3.1.1. Survey Design and Results Summary

Details on the use of in situ gamma spectroscopy for the survey of this area are provided in Reference A-2. Of interest to this report are the MDCs achieved during this survey. The average MDC was 0.367 pCi/g (0.014 Bq/g) for Cs-137 and 0.291 (0.011 Bq/g) for Co-60 with the detector located 3 meters above the ground (28 m^2 field of view). With the detector located 2 meters above the ground, the average MDC was 0.274 pCi/g (0.01 Bq/g) for Cs-137 and 0.225 (0.008 Bq/g) for Co-60. These MDCs are at most 6.8 % and 15% of the required MDC for typical FSS surveys for Cs-137 and Co-60 respectively.

A.4. References for Sections A-1 and A-2

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APPENDIX B. EXAMPLE CALCULATIONS FOR BASE CASE AND OPERATIONAL DCGLS

B.1. Connecticut Yankee Experience for Land Areas Demo

The discussed above, Connecticut Yankee (CY) expressed the total dose contribution for land areas to be from three components, the dose contribution from soil, the dose from existing groundwater and the dose from future groundwater. The following example from a CY Final Status Survey Plan illustrates how this can affect a final status survey (FSS) that addresses only the dose contribution due to soil. The dose contribution from the other pathways (i.e., existing, and future groundwater) is described in the basis for the Operational Derived Concentration Guideline Level (DCGL).

Characterization surveys and other information were used by CY to determine which area of the site is affected by existing and future groundwater. Additionally, prior to the performance the FSS on these affected areas, CY determined the values to be used for existing and future dose contributions from groundwater and the maximum allowable dose due to residual contamination which will result in satisfaction of all regulatory requirements.

B.1.1 Basis for Determining the Operational DCGL:

The DCGLs presented in the CY LTP were developed for exposures from three components; residual radioactivity in soil, existing groundwater radioactivity, and future groundwater radioactivity from the burial of remaining concrete building foundations or footings that contain residual radioactivity. Equation B-1 shows the mathematical relationship between the three components and the total dose.

(Equation B-1)

$$H_{Total} = H_{Soil} + H_{ExistingGW} + H_{FutureGW}$$

Although the total dose under the CY LTP criteria would be 25 mrem/yr TEDE from all three components, the allowable total dose under the Connecticut Department of Environmental Protection (CTDEP) radiological remediation standard for CY was 19 mrem/yr TEDE. To satisfy both the LTP and CY CTDEP criteria, the dose from soil must be reduced when using the existing and future groundwater dose values discussed above.

For this example, the survey area is assumed to be affected by existing groundwater but not future groundwater (i.e., no contaminated concrete is present in the survey area). CY had determined that the dose contribution from existing groundwater was bounded by 2 mrem/yr TEDE.

Substituting into Equation B-1:

(Equation B-2) 19 mrem/yr Total = 17 mrem/yr Soil + 2 mrem/yr Existing GW + 0 mrem/yr Future GW

The allowable dose for soil in this example survey area is 17 mrem/yr TEDE as shown by Equation B-2 above. The concentrations of residual radioactivity resulting in 17 mrem/yr TEDE are designated as Operational DCGL(s) and have been established for the radionuclides of concern as provided in Table A-1. The Operational DCGL(s) are considered the action level(s) for this example.

Table B-1 – Radionuclide Specific Base Case Soil DCGLs, Operational DCGLs and Required Minimum Detectable Concentrations (MDCs)

Radionuclide (1)	Base Case Soil DCGL (ρCi/g) (2)	Operational DCGL (ρCi/g) (3)
Ag-108m	7.14E+00	4.86E+00
Am-241 (4)	2.58E+01	1.75E+01
C-14	5.66E+00	3.85E+00
Cm-243/244	2.90E+01	1.97E+01
Co-60	3.81E+00	2.59E+00
Cs-134	4.67E+00	3.18E+00
Cs-137	7.91E+00	5.38E+00
Eu-152	1.01E+01	6.87E+00
Eu-154	9.29E+00	6.32E+00
Eu-155	3.92E+02	2.67E+02
Fe-55	2.74E+04	1.86E+04
H-3	4.12E+02	2.80E+02
Mn-54	1.74E+01	1.18E+01
Nb-94	7.12E+00	4.84E+00
Ni-63	7.23E+02	4.92E+02
Pu-238	2.96E+01	2.01E+01
Pu-239/240	2.67E+01	1.82E+01
Pu-241	8.70E+02	5.92E+02
Sr-90	1.55E+00	1.05E+00
Tc-99	1.26E+01	8.57E+00

(1) Bold indicates those radionuclides considered Hard to Detect (HTD)

(2) The Base Case Soil DCGL(s) are specified by the CY LTP and are equivalent to 25 mrem/yr TEDE

(3) The Operational DCGL is equivalent to achieving 17 mrem/yr TEDE

(4) Americium-241 can be analyzed by gamma and alpha spectroscopy and is Easy to Detect (ETD). The preferred result is by alpha spectroscopy when both analyses are performed.

APPENDIX C. CROSSWALK BETWEEN LICENSE TERMINATION PLAN AND NUREG-1700

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APPENDIX D. SUGGESTED FEDERAL AND STATE REGULATORY INTERFACE PLAN

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APPENDIX F. SITE SPECIFIC DOSE MODELING EXPERIENCES

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F.1.1 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 6. 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 F-2) 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).

F.1.2 Dose Assessment Model – Groundwater

As discussed above, the DCGLs for soil at CY were determined with the assumption that no groundwater contamination existed to account for groundwater contamination present at the time of license termination (as CY 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 cm3/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 "User's Manual for RESRAD Version 6.0," (Reference F-3) from which the following equation was derived:

$$c = \frac{1000 S_o \rho_b}{\left[1 + \left(K_d \rho_b / n\right)\right] n}$$

Equation F-2

where,

C = Equilibrium groundwater concentration (pCi/l)

S_0 = Initial principal radionuclide concentration in contaminated zone (pCi/g)

ρ_b = Bulk density of contaminated zone (g/cm3)

K_d = Distribution Coefficient of contaminated zone (cm3/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.

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F.2. Big Rock Point Experience

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. The site specific DCGLs, developed using the modified Resident Farmer Scenario 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.

F.3. Rancho Seco Experience

As discussed above, the Sacramento Municipal Utility District (SMUD) does not plan to transfer the Rancho Seco Nuclear Generating Station property 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 F-4).

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 include 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.

F.4. Parameter Sensitivity Analysis

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. 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 F-3) 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 F-5), 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 F-6).

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 F-6).

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 F-7) 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 F-8) 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).

In NUREG 1757 (Reference 9) Appendix Q, the NRC recommends increased justification to use the literature probability density functions (PDFs) such as those in Tables 12.13.1 – 12.13.5 of the Argonne National Lab, Data Collection Handbook (DCH Reference F-15) in the sensitivity analysis used to determine if the Kd for the radionuclides of concern at a particular site is sensitive to the dose to the future user of the site.

The NRC has stated in the response to comments and questions on NUREG 1757, Volume 2, Revision 2 concerning this topic (Reference F-17):

> *NRC responded that the issue with use of literature values for Kds is that data available to develop the parameter distributions could be based on (i) sparse data, or (ii) reflect a range of sites with 25th or 75th percentile values not necessarily being with the range of values for any particular site. Additional information in the form of site-specific information on soil/mineral types, groundwater chemistry and quality could be used to support the parameter values if the parameter is found to be risk-significant and sufficient information is available in the literature to support the parameter value for a particular set of site-specific conditions (i.e., site-specific field or laboratory experiments are not always needed to support the site-specific parameter value even if found to be risk-significant). Radionuclides contributing less than 10 percent of the dose*

standard are considered insignificant and additional support for these radionuclides would not be required. Depending on the radionuclides of concern and the importance of the parameter value to dose, a graded approach will be used to determine the need for additional site-specific support. Licensees should work with NRC reviewers early in the process to determine the need for additional site-specific support for risk-significant parameters such as Kd

In the development of recent License Termination Plans, Kd has not been shown to be a sensitive input parameter in dose modeling for the predominant radionuclides of concern (Co-60 and Cs-137 and to a lesser extent H-3 and Sr-90) at nuclear power plant sites. For example:

- For the Zion plant, Kd was not a sensitive input parameter for any of the radionuclides of concern.

- For the La Crosse plant, Kd was a sensitive input parameter for only Ni-59, N-63, Tc-99 and Np-237.

The four radionuclides listed for La Crosse are typically not detected or detected in insignificant quantities in soils at nuclear power plants. For an individual plant, site characterization data may likely confirm that these types of radionuclides are not present in soil in quantities to be a significant dose contributor. If this is case, the licensee should propose to the NRC that increased justification for the use of DCH Kd probability density functions is not required for these radionuclides.

F.4.1 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. 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). 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 F-6) 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.

F.4.2 Applicability of Surface Soil DCGLs to Sub-Surface Soil

As discussed above, 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.

F.4.3 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 as an alternate but not a compliance scenario. 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 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.

F.5. Calculated Dose 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 that the dose from the maximum allowable concentrations allowed by the Industrial Worker Scenario will be 25 mrem/yr approximately 30 years after the projected release of the site when using the Resident Farmer Scenario.

It is highly unlikely that SMUD would consider transferring 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. The reduction in dose from residual radioactivity over time is noteworthy for operating plants and permanently shut down sites that are in a SAFSTOR mode (i.e., to be decommissioned later). 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 these results are 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.

[redacted]

F.7. Structures

F.7.1 Overview of Options for the Development of Site Release Limits for Structures

The evolution of site release limits for buildings was like that described in Chapter 6 for land areas. The early 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.

[redacted]

F.7.2 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.

F.7.3 NRC Published Screening Values for Structures

The NRC has published screening values for residual radioactivity in structures. These screening values are provided in 63 FR 64132, “Supplemental Information on the Implementation of the Final Rule on Radiological Criteria for License Termination” (Reference F-10). These 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 a 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 9) are as follows:

- Contamination on building surfaces should be surficial and non-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 9) 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.

F.7.4 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 Reference F-1.

F.7.5 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.

F.7.6 Building Occupancy Scenario

The conceptual model underlying Building Occupancy Scenario dose model for CY consisted of a room of fixed area [10m by 10m by 2.5m high], uniform concentrations of residual radioactivity on all room surfaces, and the receptor located at the center of the room at a height of 1m. 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 F-11). Volumetric sources consisted of concrete to the depth of 0.305m (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.

This scenario assumes that the building will be used as an office building after the 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 was used for the Building Occupancy DCGL determination.

[redacted]

F.7.7 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.

Table F-3
CY DCGLs for Building Demolished (Concrete Debris Scenario)

Radionuclide	Concrete Debris DCGLs(pCi/g)	DCGL for Volumetric Sources (Building Occupancy Scenario(pCi/g)
H-3	9.05E+01	1.47E+03
C-14	2.05E+01	1.18E+08
Mn-54	5.51E+01	9.06E+00
Fe-55	8.96E+01	9.54E+07
Co-60	9.07E+01	2.90E+00
Ni-63	1.29E+02	4.11E+07
Sr-90	3.77E-01	2.38E+03
Nb-94	7.74E+00	4.83E+00
Tc-99	2.85E+01	3.09E+07
Ag-108m	2.59E+01	4.84E+00
Cs-134	3.21E+02	4.93E+00
Cs-137	6.45E+02	1.37E+01
Eu-152	2.27E+02	6.70E+00
Eu-154	1.94E+02	6.11E+00
Eu-155	9.53E+03	3.23E+02
Pu-238	1.14E+01	6.61E+02
Pu-239	1.00E+01)	6.02E+02
Pu-241	1.49E+02	3.12E+04
Am-241	4.42E+00	4.16E+02
Cm-243	3.83E+00	7.53E+01

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 F-3 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.

F.7.8 Zion Experiences

The Zion site also modeled the dose from concrete debris that is to be used as backfill on-site. Zion demonstrated that all concrete designated as backfill material in basements is clean through the Unconditional Release Survey (URS) program at Zion. Materials unconditionally released from Zion, regardless of their point of origin on the site, have been verified to contain no detectable plant-derived radioactivity and are free to be used and relocated anywhere offsite without tracking, controls, or dose considerations (Reference F-12).

Although the concrete debris to remain onsite and used as clean fill can be viewed as having a no dose impact, a dose value will be assigned for the purpose of demonstrating compliance with 10 CFR 20.1402 in the same manner as other materials to remain at license termination that are surveyed and found to not contain detectable activity. The “detection limit” used for the dose calculation is conservatively assumed to be the maximum scan MDC of 5,000 dpm/100 cm^2 allowed in the URS program. Actual detection limits in the unconditional release program were lower than this value.

The vast majority of clean concrete fill to be used came from five buildings: Containment, Turbine, Crib House/Forebay, Service Building and Interim Waste Storage Facility. Because the concrete was from both Containment and other structures, the dose calculation was performed using both the Containment and Auxiliary ROC mixtures. Consistent with the bounding approach used for the clean concrete assessment, the Containment mixture was applied to all concrete. In addition, when applying the ROC mixture, the 5,000 dpm/100 cm2 maximum detection limit was assumed to be 100% Cs- 137. The remaining radionuclide concentrations were added to the Cs-137 concentration at their respective ratios to Cs-137.

The dose values are calculated separately for each basement assuming that the entire basement void is filled with concrete only. This conservatively includes the top three feet of fill which will be soil for all basements and not concrete. The total dose results for each basement, assuming a scan MDC value of 5,000 dpm/100 cm2 and including all ROC, are provided in Table F-4.

Table F-4
Dose Assigned to Clean Concrete Fill at Zion

Basement	Auxiliary	Containment	SFP/Transfer Canal	Turbine	Crib House/ Forebay	WWTF
Dose (mrem/yr)	9.94E-01	1.77E+00	1.52E-01	1.58E+00	1.57E+00	6.40E+00

F.7.9 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 Equation 6-1, the CY compliance equation. As discussed above, the “Future Groundwater” component might not apply in all situations. Connecticut Yankee submitted the Basement Fill Model for NRC approval as a "realistic" scenario.

Also, as discussed above, NRC encouraged the use of realistic scenarios, where appropriate, in its updated guidance. CY made the decision, in 2004, to dispose of all 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 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 (BFM) 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 F-1) 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 travel 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 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.

- 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 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.

[redacted]

[redacted]

F.7.10 Zion Experience with Basement Fill Model

The Zion site also utilized the BFM to assess basements to remain after decommissioning in a similar fashion to CY. The BFM calculates the annual dose to the AMCG from surface and volumetric residual radioactivity remaining in the basement piping and structures listed in the following bullets:

- Concrete in the Containment Buildings
- Auxiliary Building Turbine Building Concrete
- Crib House and Forebay Concrete
- Wastewater Treatment Facility Concrete
- Spent Fuel Pool Concrete
- Main Steam Tunnels (Unit 1 and Unit 2) Concrete
- Circulating Water Intake Piping Steel Pipe (Site and Lake)
- Circulating Water Discharge Tunnels

The End State Basements comprised of steel and/or concrete structures, which were covered by at least three feet of clean soil and physically altered to a condition which would not realistically allow the remaining structures, if excavated, to be occupied. The exposure pathways in the BFM are associated with residual radioactivity in floors and walls that is released through leaching into water contained in the interstitial spaces of the fill material. For Zion, the BFM assumes that the inventory of residual radioactivity in a given building is released either instantly or over time by diffusion, depending on whether the activity is surficial or volumetric, respectively.

The activity released into the fill water will adsorb onto the clean fill, as a function of the radionuclide-specific distribution coefficients, resulting in equilibrium concentrations between the fill and the water. Consequently, the only potential exposure pathways after backfilling, assuming the ‘as-left’ geometry, are associated with the residual radioactivity in the water contained in the fill.

A water supply well is assumed to be installed within the fill of the basement. The well water is then used for drinking, garden irrigation, pasture/crop irrigation, and livestock water supply in the Resident Farmer scenario.

The BFM is implemented using two computational models. The Disposal Unit Source Term - Multiple Species (DUST-MS) model is used to calculate the maximum water concentrations in the fill material of each basement for a given inventory of residual radioactivity (pCi/L per mCi).

The RESRAD v7.0 model is used to determine the dose to the Resident Farmer as a function of the water concentration (mrem/yr per pCi/L). BFM Groundwater (GW) Dose Factors are then calculated for each

Basement and each Radionuclide of Concern (ROC) by combining the results of the two models with units of mrem/yr per mCi total inventory.

Basement DCGLB values were calculated for each of the Basements listed in the bullets above. The Circulating Water Discharge Tunnels were accounted for by adding the surface area (and corresponding source term) to the Turbine Basement during the DCGL calculation. The Circulating Water Intake Piping was accounted for by adding the surface area to the Crib House/Forebay Basement during the DCGLB calculations. Therefore, the DCGLB values calculated for the Turbine Basement also apply to the Circulating Water Discharge Tunnels and the DCGLB values for the Crib House/Forebay also apply to the Circulating Water Intake Piping.

The Steam Tunnel surface area and volume were included with the Turbine Basement in the calculation of BFM Dose Factors and DCGLs. The Turbine Basement DCGLB values therefore also apply to the Steam Tunnel.

F.8. Rancho Seco Experience

As discussed above, 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.

F.8.1 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 F-11). RESRAD-BUILD Version 3.3 (released summer 2005 by Argonne National Laboratory) 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.

Table F-4
Comparison of Rancho Seco Building Surface DCGLs 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

F.8.2 Building Renovation/Demolition Scenario

The building renovation/demolition scenario, as described in NUREG/CR- 5512, Vol. 1 (Reference F-11) along with the input data template and input parameter values provided in ANL/EAD/03-1 (Reference F-13) 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 F-4 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.

F.8.3 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 F-4 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.

F.9. Buried Piping

F.9.1 Overview of Buried Piping Dose Modeling

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. This section addresses experiences with the development of site release limits for buried piping.

F.9.2 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 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 Equation F-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 contaminated grout.

Consistent with these simplified assumptions, the Derived Concentration Guidance Levels (DCGLs) calculated for the Concrete Debris Scenario (Table F-3) were applied to determine the dose from the slug of grout. To calculate the release limits for the piping (corresponding to 1 mrem/yr), first, the portion of the 25 mrem/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 (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.

F.9.3 Rancho Seco Experience

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.

F.10. Embedded Piping

F.10.1 Overview of Embedded Piping Dose Modeling

EPRI Report 1000908, "Remediation of Embedded Piping, Trojan Nuclear Plant Decommissioning Experience," (Reference F-14), provides a detailed discussion of experiences with embedded piping at the Trojan Plant. The following are the highlights of that report.

F.10.2 Trojan Plant Experience

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 thought to be more economical to leave this piping in place if it was shown to meet the limits described below. F-14 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.

F.10.3 Dose Modeling Approach for Embedded Piping

Trojan decided to allot a portion of the 25 mrem/yr unrestricted release criteria for buildings 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 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.

F.10.4 Maine Yankee Experience

The dose due to contamination in 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 operational DCGLs 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.

F.10.5 Rancho Seco Experience

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 to ensure compliance with the annual dose limit of 25 mrem/yr The MicroShield® computer code was used to evaluate dose from embedded piping.

F.10.6 Zion Experience

For the Zion site, embedded pipe is defined as a pipe that runs vertically through concrete walls or horizontally through concrete floors and is contained within a given building. Additionally, the release pathway for the residual radioacticvity in embedded piping for Zion was considered to be into the basement where the piping is contained. The dose from the embedded piping was summed with the dose from the wall and floor surface of the basement that contains the embedded pipe. A DCGL in units of pCi/m^2 was calculated for each embedded pipe survey unit in given the designation $DCGL_{EP}$. To eliminate the potential for activity in embedded pipe to result in the release of radioactivity that could potentially result in higher concentrations than predicted by the Basement Fill Model used for Zion, remediation and grouting action levels were established. However, the dose from embedded pipe was calculated using the $DCGL_{EP}$ values in order to accurately account for the dose.

The embedded pipe survey units are listed in Table F-5 along with the total internal survey area of the pipes in the surveuy unit. The IC-sump embedded pipe is very limited with a total surface area of 1.05m^2 for each Unit 1 and Unit 2. To provide a reasonable maximum value for the DCGL, a nominal value of 100m^2 was assumed for the surface area of the IC sump embedded pipe survey unit. The U2 Steam Tunnel surface area was slightly lower that the U1 area (46.88 m^2 versus 46.39 m^2). For simplicity, the higher, more conservative area was applied to both Steam Tunnel Floor Drain DCGL calculations. For more details on the calculation of the $DCGL_{EP}$ see the Zion License Termiantion Plan (Reference F-12).

Table F-5
Embedded Pipe Survey Unit Surface Areas

Embedded Pipe	EP SU Surface Area (m^2)
Auxiliary Floor Drains	299.41
Turbine Floor Drains	302.43
U1 Containment IC-Sump Drain	1.05 (100) [(1)]
U2 Containment IC-Sump Drain	1.05 (100) [(1)]
U1 Steam Tunnel Floor Drain	46.88[(2)]
U2 Steam Tunnel Floor Drain	46.88[(2)]
U1 Tendon Tunnel Floor Drain	51.41
U2 Tendon Tunnel Floor Drain	51.41

(1) The total surface area of unit 1 and unit 2 IC Sump Drains are 1.05 m^2 each. To provide a reasonable maximum value for the DCGL a nominal area of 100 m^2 was assumed for the DCGL calculation.
(2) Higher surface area applied to both U1 and U2 Steam Tunnel Floor Drains. U2 area is 46.39 m^2.

F.11. References for Appendix F:

F-1 Haddam Neck Plant License Termination Plan, Revision 4, November 2006

F-2 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).

F-3 Argonne National Laboratory, ANL/EAD-04, “User’s Manual for RESRAD Version 6,” July 2001.

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

F-5 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

F-6 Argonne National Laboratory, NUREG/CR-6697, “Development of Probabilistic RESRAD 6.0 and RESRAD-BUILD 3.0 Computer Codes”, December 2000

F-7 U.S. Nuclear Regulatory Commission, NUREG/CR-6692, “Probabilistic Modules for the RESRAD and RESRAD-BUILD Computer Codes,” November 2000

F-8 U.S. Nuclear Regulatory Commission, NUREG/CR-6676, “Probabilistic Dose Analysis Using Parameter Distributions Developed for RESRAD and RESRAD-BUILD Computer,” May 2000.

[redacted]

F-10 Federal Register, 63 FR 64132, Supplemental Information on the Implementation of the Final Rule on Radiological Criteria for License Termination, dated November 18, 1998.

F-11 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.

F-12 Zion Station Restoration Project License Termination Plan, Chapter 6, Revision 2

F-13 Argonne National Laboratory, ANL/EAD/03-01, “User’s Manual for RESRAD-BUILD Version 3,” June 2003.

F-14 EPRI Report # 1000908, “Remediation of Embedded Piping, Trojan Nuclear Plant Decommissioning Experience”, dated October 2000.

F-15 Argonne National Laboratory, ANL/EVS/TM-14/4, “Data Collection Handbook to Support Modeling Impacts of Radioactive Material in Soil and Building Structures,” 2015

F-16 “Data Collection Handbook to Support Modeling Impacts of Radioactive Material in Soil and Building Structures,” ANL/EVS/TM-14/4, 2015 (DCH 2015)

F-17 NRC Memorandum, C. McKenney to C. Barr, Summary of October 11, 2022, Hybrid Information Meeting on NUREG-1757, Volume 2, Rev. 2, Guidance Updates, dated October 24, 2022.

APPENDIX G. EXAMPLE OF CHARACTERIZATION, REMEDIATION, AND FINAL STATUS SURVEY OF GROUNDWATER

The following example shows how one plant carried out this process and achieved license termination when groundwater contamination was present at the site.

Assessments of any residual activity in groundwater are generally conducted using groundwater monitoring wells. Due to the presence of extensive groundwater contamination at Connecticut Yankee, the monitoring well experience at CY provides a good example of the measures needed to address this dose pathway.

Remediation of soil at Connecticut Yankee to meet the site release criteria proved to be a challenging and lengthy effort. A summary of the steps needed to address groundwater contamination at CY is described in the following:

G.1. Initial Groundwater Characterization

The site characterization at CY identified areas where groundwater contamination was suspected. CY, working with the state of Connecticut Department of Environmental Protection (CT DEP), initiated a Phase 1 Groundwater Monitoring Program. As part of this program, CY installed groundwater monitoring wells in various areas of the site in 1997 and began monitoring those wells in late 1997.

Wells were in three areas of the plant:

- The industrial area of the plant where the reactor containment and the building housing the auxiliary equipment (Primary Auxiliary Building) were located.
- The Peninsula Area located between the discharge canal and the Connecticut River
- The landfill area where trace levels of radionuclides had been identified during their Historical Site Assessment and subsequent site characterization.

Per the program developed with the CT DEP, groundwater samples were analyzed for tritium, gamma radioisotopes along with additional analyses for Gross Alpha, and Gross Beta groundwater concentrations. H-3 is an excellent tracer radionuclide as it is the most mobile of the radionuclides, moving through soil and bedrock like water to which it is essentially identical. Although H-3 levels were elevated during initial sampling, concentrations dropped quickly as natural attenuation occurred. Other radionuclides are slowed in their movement through soil and the fractures in bedrock as they chemically interact with the granular material present.

G.2. Detailed Groundwater Related Characterization

After initial characterization information had been collected, a more refined groundwater monitoring program was developed, again working with the CT DEP. This program was called the Phase 2 Hydrogeologic Work Plan and was approved in May 2001.

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