TECHNICAL REPORT

NEI 22-01, Rev. 2

License Termination Process

Prepared by the Nuclear Energy Institute
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Revision Table

Revision	Description of Changes	Date Modified	Responsible Person

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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 can 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 pre-2019 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) 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 level of detail in reporting survey results is a major undertaking and has been the source of significant project delays. Delays have also occurred in obtaining initial approval of the LTP due primarily to factors such as:

- An inadequately detailed site characterization
- Obtaining concurrence on the dose modeling approach to be used
- Level of detail in the FSS approach and methods described in the plan

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 and have not proven to be particularly helpful to the typical commercial nuclear power reactor licensee.

In addition, the NRC's guidance does not satisfactorily address a few technical subjects that have emerged as major 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.

Experience has highlighted the importance of a high level of coordination and communication between the licensee, the NRC region, and NMSS headquarters staff both before submittal of the LTP and throughout its review by NRC staff.

Objectives

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:

- How to navigate the many regulatory documents that provide decommissioning guidance, and direct the users' attention to those requirements, practices and recommendations that are most applicable to the typical commercial reactor preparing for or undergoing decommissioning.
- Propose 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.
- Provide 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.
- Provide technical solutions for areas not currently addressed in NRC or EPA regulatory guidance.
- Provide 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.

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Table of Equations, Figures, and Tables

1 INTRODUCTION TO THE LICENSE TERMINATION PROCESS

1.1 Purpose and Scope of NEI 22-01

The purpose of this document is to provide information that assists nuclear power plant licensees in the preparation of License Termination Plans (LTPs). The LTP is an important document that is approved by the NRC and guides the process of decommissioning the plant and achieving termination of the NRC license. There are numerous NRC regulations and guidance documents that are applicable to the site release process at a nuclear power plant. This document intends to state and summarize the important requirements and guidance from the applicable NRC documents. Additionally, selected experiences from past LTPs will be provided as examples for future use by licensees. The following is a listing of the NRC regulations and guidance documents that are applicable to the site release process along with a high-level summary of the content of these documents.

1.2 NRC Regulatory Requirements

1. Title 10 of the *Code of Federal Regulations* (10 CFR) Part 20, "Standards for Protection Against Radiation," Subpart E, "Radiological Criteria for License Termination," requires licensees to demonstrate compliance with radiological criteria for license termination, including use of radiological surveys to show that release criteria are met to reach favorable decisions regarding license termination.

2. 10 CFR 20.1402, "Radiological criteria for unrestricted use," contains requirements for licensees to demonstrate that a site will be considered acceptable for unrestricted use.

3. 10 CFR 20.1403. "Radiological criteria for restricted use," contains requirements for licensees to demonstrate that a site will be considered acceptable for restricted use.

4. 10 CFR 20.1501, "General," requires licensees to demonstrate that residual radioactivity, including existing groundwater, has been adequately characterized.

5. 10 CFR 50.82, "Termination of license," paragraph 50.82(a)(9) outlines the requirements for submitting a license termination plan. It requires the license termination plan include site characterization and an updated site-specific decommissioning cost estimate that includes an estimate of the cost of remaining decommissioning work.

6. 10 CFR 50.83, "Release of part of a power reactor facility or site for unrestricted use," outlines the requirements for release of part of a power reactor facility or site for unrestricted release before termination of the license including the requirement to perform adequate surveys to demonstrate compliance with radiological criteria for unrestricted use specified in 10 CFR 20.1402, "Radiological Criteria for Unrestricted Use," for impacted areas (i.e., areas that have some potential for radionuclide contamination are classified as impacted areas).

7. 10 CFR 50.59, "Changes, tests and experiments," contains requirements for the process by which licensees may make changes to their facilities and procedures as described in the safety analysis report, without prior NRC approval, under certain conditions.

8. 10 CFR 51.45, "Environmental report," describes the requirements related to a licensee's environmental report. It provides the requirements associated with the status of compliance with applicable environmental quality standards.

1.3 Applicable NRC Guidance

1. DUWP–ISG-02, "Radiological Survey and Dose Modeling of the Subsurface to Support License Termination," date October 2023,[1] supplements NUREG-1757, Volume 2, and provides guidance on radiological survey approaches for substructures as well as limitations of codes such as RESRAD–ONSITE in assessing groundwater dependent pathway doses for submerged sources such as reactor basement substructures.

2. NUREG-1575, "Multi-Agency Radiological Survey and Site Investigation Manual" (MARSSIM),"[2] guidance focuses on the demonstration of compliance during the final status survey following scoping, characterization, and any necessary remedial actions. It outlines recommended survey coverage for both structures and land areas.

3. NUREG-1576, "Multi-Agency Radiological Laboratory Analytical Protocols [MARLAP] Manual," dated July 2004,[3] provided definitions for critical level and minimum detectable concentration (or activity) on which to base detection decisions for water samples.

4. NUREG-1700, "Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans," dated April 2018,[4] ensures the quality and uniformity of NRC staff reviews and to present a well-defined base from which to evaluate the requirements for terminating the license of a nuclear power plant. Appendix B, "LTP Areas That Cannot Be Changed Without NRC Approval," outlines LTP areas that cannot be changed without NRC approval, including those that require Commission approval under 10 CFR 50.59.

5. NUREG-1748, "Environmental Review Guidance for Licensing Actions, Associated with NMSS [Office of Nuclear Material Safety and Safeguards] Programs," dated August 2003,[5] provides general procedures for the environmental review of licensing actions that support licensees when preparing environmental reports for submission to the NRC.

6. NUREG-1757, "Consolidated NMSS Decommissioning Guidance – Characterization, Survey, and Determination of Radiological Criteria, Volume 2," Revision 2, dated July 2022,[6] provides guidance on radiological surveys and dose modeling to develop cleanup criteria to support licensees in preparing decommissioning plans, LTPs, FSSs, and other technical decommissioning reports for NRC submittal. The NRC staff also uses this guidance in reviewing these documents and related license amendment requests (LARs).

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[1] ADAMS Accession No. ML23177A008

[2] https://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1575/index.html

[3] https://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1576/index.html

[4] ADAMS Accession No. ML18116A124

[5] ADAMS Accession No. ML032450279

[6] ADAMS Accession No. ML22194A859

7. NUREG/CR-5512, “Residual Radioactive Contamination from Decommissioning: User's Manual DandD Version 2.1,”[7] Volume 2, dated April 2001, provides a screening methodology to address the technical dose criteria contained in NRC's Radiological Criteria for License Termination rule.

8. NUREG/CR-7021, “A Subsurface Decision Model for Supporting Environmental Compliance,” dated January 2012,[8] presents a framework focused on development of a conceptual site model referred to as a contamination concern map and decision framework for conducting a subsurface compliance survey and analysis for sites that have been remediated for radioactive contamination.

9. Regulatory Guide (RG) 1.179, Revision 2, “Standard Format and Content of License Termination Plans for Nuclear Power Reactors,” dated July 2019,[9] guides the NRC staff in conducting safety reviews and assists licensees in developing an LTP. This RG includes format and technical content of an LTP submittal, including a supporting environmental report and demonstration of dose for residual radionuclides that includes the groundwater media.

10. NUREG/CR-6676, “Probabilistic Dose Analysis Using Parameter Distributions Developed for RESRAD and RESRAD-BUILD Codes,”[10] for demonstration of compliance with the dose/risk criteria, as well as several documents the NRC has listed in the NUREG for NEI consideration.

11. NUREG/KM-0016, “Be risk SMART: Guidance for Integrating Risk Insights into NRC Decisions,”[11] which provides a systematic approach to making risk-informed decisions across disciplines.

1.4 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, 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. NUREG-1757, Volume 2, Revision 2, Section G.3.2, emphasizes the need for FSS of open surfaces (e.g., excavations or substructures) to be performed prior to backfill due to the difficulty in accessing the surfaces after backfilling.

Additional draft NRC guidance is provided in DUWP-ISG-02 (Reference 5-6). including lessons learned from inadequate survey of open surfaces in the subsurface, misapplication of clean-up levels,

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[7] https://www.nrc.gov/reading-rm/doc-collections/nuregs/contract/cr5512/

[8] https://www.nrc.gov/reading-rm/doc-collections/nuregs/contract/cr7021/

[9] ADAMS Accession No. ML19128A067

[10] https://www.nrc.gov/reading-rm/doc-collections/nuregs/contract/cr6676/

[11] https://www.nrc.gov/reading-rm/doc-collections/nuregs/knowledge/km0016/index.html

inadequate depth of sample, and the lack of opportunity for confirmatory survey. Use of this document can provide additional detail on the level of information needed to support FSS for backfill of open surfaces in the subsurface including excavations and substructures.

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. In particular, conducting these surveys may require that operational DCGLs be used to account for multiple dose pathways as discussed in Section 5 and Appendix B. The parameters that drive the operational DCGL values may not be known prior to LTP approval. Therefore, these decisions must be made with care, ample conservatism and margin.

Additional examples of work that was performed at risk with the associated site release implications are given in Appendix I.

1.4.1 Communications among Licensee and Regulators

Decommissioning licensees should begin discussions with NRC and state regulators 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. The licensee's goal should be to gain NRC's acceptance of the document for review as soon as possible after submittal. Multiple meetings are encouraged prior to submittal to discuss the approach and technical detail of each LTP chapter. For example, Chapter 2, "Site Characterization," is a prime candidate for early discussion (See Appendices D & E). A suggested approach is to conduct a series of detailed pre-acceptance review meetings to discuss the content and technical approach being proposed for each LTP chapter.[12]

Early agreement should also 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 LTP should be submitted in its entirety along with the key supporting technical reports (e.g., Historical Site Assessment, Site Characterization Report, etc.) referenced in the LTP. The LTP submittal is a complex license amendment request and should be submitted earlier than 2 years prior to requesting release of the site. The approved LTP is incorporated by reference into the FSAR (DSAR) and is required to be updated per 10 CFR 50.71 if the Part 50 license is retained for the ISFSI and any remaining property. Appendix E provides how the LTP submittal, NRC review/approval and use throughout the decommissioning can fit into the overall decommissioning.

During actual decommissioning, remediation, and survey activities, periodic meetings should be planned to discuss LTP progress, results of NRC Regional interface/inspections, MARSSIM (Reference 1-1) 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

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[12] See Pilgrim Preapplication Readiness Assessment Plan, ML24129A104

regulators is provided in Appendix D, and a typical timeline for license termination-related activities and associated regulatory interactions is provided in Appendix E.

1.4.2 Standard Format and Content

Standard format and minimal content for the LTP is found in Regulatory Guide 1.179 (Reference 1-2). 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.

1.4.3 Acceptance Criteria and Regulatory Review

The guidance for NRC LTP reviewers is contained in NUREG-1700 (Reference 1-3). Author(s) of individual chapters of the LTP should familiarize themselves with the acceptance criteria, along with the Regulatory 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.

1.4.4 Crosswalk between LTP and NRC regulations and guidance documents

Appendix C provides an example of the type of template that can be used as a crosswalk between the LTP and various NRC regulations and guidance documents. It should be noted that the crosswalk in Appendix C does not include all regulations and guidance statements applicable to the preparation of an LTP. Use of this type of template and including it in the LTP submittal will facilitate regulatory reviews.

1.4.5 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.

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 per 10 CFR 50.83:

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.

In addition, for release of non-impacted areas, the submittal request must include:

1. The results of the evaluations performed above.
2. A description of the part of the facility or site to be released.
3. The schedule for release of the property.
4. The results of the evaluations performed in accordance with § 50.59.
5. A discussion that provides the reasons for concluding that the environmental impacts associated with the licensee's proposed release of the property will be bounded by appropriate previously issued environmental impact statements.

For release of impacted areas, the licensee must submit a license amendment application for the release of the property. The application must include:

1. The information specified in paragraphs (b)(1) through (b)(3) of 10 CFR 50.83;
2. The methods used for and results obtained from the radiation surveys required to demonstrate compliance with the radiological criteria for unrestricted use specified in 10 CFR 20.1402; and
3. 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.
4. A reason why the impacted area needs to be removed from the license before the LTP is approved

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.

Partial Site Release Process

Using the experience gained from previous partial site release requests, the following process is recommended in achieving a partial site release at the project site.

Phase 1 – Redefining the Licensed Site Boundary

Phase 1 entails redefining the licensed portion of the site. A review of the HSA and associated reference documents, and consulting with personnel familiar with past, current, and planned future uses of the site is needed. Additionally, the locations and size of areas needed to support decommissioning activities such as building demolition, rail infrastructure, and waste storage/packaging should be determined. It is also critical that any areas released from the license have no potential for negative impact from future decommissioning activities. Consideration was also given to include, in the remaining licensed site, the areas that encompass the estimated 25 mR/year zone surrounding the ISFSI. Areas that will remain under the Part 50 License are of course the footprint of the Radiologically Restricted Area (including the ISFSI) and any other areas that may be needed to accommodate DandD activities and infrastructure such as:

- laydown areas for equipment and large components
- waste container storage/packaging
- recommended rail spur including gondola car storage/loading
- areas to be used for the storage and potential size reduction of large components

The remaining areas of the site that are potentially within the scope of a partial site release (whether classified as impacted or non-impacted) are sub-divided into appropriately defined survey units to better manage the surveys.

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. Because a partial site release in this case would change the site as licensed, a license amendment application for the proposed partial site release would be required regardless of the amount of residual radioactivity present in the area to be released. Also, a request for a partial site release of an impacted area will require a license amendment and a reason for why the impacted area needs to be removed from the license before the LTP is approved. The license amendment request will also require a supplement to the environmental report under 10 CFR 51.53 (e.g., LTP Chapter 8) to be included with the request.

Phase 2 - Preparation of Program Documents

Phase 2 primarily entails the preparation of program documents to standards that ensure the program content, implementation and reporting of results consists of the quality and pedigree that meet NRC requirements. A list of the generally required upper-tier documents and implementing procedures, and a summary of the purpose of each, is provided below:

- A Quality Assurance Project Plan (QAPP)

  The QAPP serves to ensure that site characterization is performed using approved written procedures by trained individuals and properly calibrated instruments that are sensitive to the potential contaminants of concern. The QAPP also describes the quality assurance requirements and quality controls needed for sampling and analytical methodologies which limit the introduction of errors into analytical data required to support the partial release of the site for unrestricted use in accordance with NRC requirements.

- A Radiological Survey (Characterization) Plan for Partial Site Release

  For impacted areas, the Characterization Plan presents the approach and process to be used for the characterization of the site. It should be developed to provide guidance and direction to personnel responsible for implementing and executing release (characterization) survey activities. The characterization plan further works in conjunction with implementing procedures and survey unit specific survey instructions (survey packages) that are developed to acquire the requisite characterization data.

- Characterization Survey Package Development

  The purpose of this procedure is to provide instructions for the development, implementation and review of characterization survey plans and sampling instructions. The survey design process should incorporate the Data Quality Objectives (DQO) process described in MARSSIM.

- Characterization Survey Data Assessment

  This procedure describes the Data Quality Assessment (DQA) process used to evaluate characterization survey data and verify the results are below the established release criteria.

- Sample Media Collection for Site Characterization

  The purpose of this procedure is to provide instructions on obtaining soil, asphalt, sediment, concrete, stone, liquid, vegetation, and other sample media for use in characterization surveys.

- Sample Media Preparation for Site Characterization

  This procedure provides instruction for the preparation of bulk material samples for radiological analysis.

- Instrumentation Use and Control

  Procedures are developed to provide instructions on the use of the various instruments used during the release surveys and any measures necessary for controlling the instruments while in the field.

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.” (Reference 1-4) It should be noted that these documents may be used with minimal revision in support of the future characterization of land, system and structure survey units that remain in the license.

<u>Phase 3 – Survey Implementation for Partial Site Release</u>

For impacted areas, Phase 3 entails the implementation phase of the project including survey design, survey performance and the reporting of survey results. Radiological surveys will be performed within land areas and structures that have been earmarked for partial site release. A sample plan will be developed for a background study and for each survey unit during the release surveys.

Non-impacted areas can be released based on historical data (i.e., no radiological surveys are required). However, scoping and characterization surveys may be needed initially in order to justify the non-impacted classification.

A background study will be performed within a non-licensed portion of the site (or an offsite location with similar geographical attributes) to ascertain the background radiological concentrations in soil and will support performance of a Wilcoxon Rank Sum (WRS) Test, Kruskal-Wallis Test, or other non-parametric statistical tests, if necessary, as well as verifying non-impacted classifications. The background study will consist of obtaining bulk material samples of the various media that are found on site and the results will be documented in a Technical Support Document (TSD) to eventually be included as part of Chapter 2 of the License Termination Plan (LTP).

For impacted areas, the release/characterization survey will generally consist of gamma scans of land areas (including soil, asphalt, concrete, sediment, water) augmented by bulk media sampling. Beta scans will be performed as necessary on structures with limited intrusive sampling performed. 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. In general, impacted survey units will have a minimum of 5% of the accessible surface areas scanned. The number of randomly selected bulk material samples required in each survey unit is estimated to be 14. In addition, QC split samples will be obtained at 5% of the sample locations. Additional judgmental samples will be obtained at areas where contamination could concentrate such as outfalls, low points, drainage ditches, etc.

All scan and sample locations will be designated on a survey map along with GPS coordinates. This facilitates returning to the survey locations for investigative or regulatory verification purposes.

1.4.6 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. Areas that require prior NRC approval are changes that:

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

- Decrease a survey unit area classification (i.e., 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 before implementing the change in classification.

- Increase the derived concentration guideline levels and related minimum detectable concentrations (MDCs) for both scan and fixed measurement methods. If MDCs are increased (relative to what was approved) the licensee should request NRC approval.

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

- Change the statistical test applied to a test other than the Sign test or Wilcoxon Rank Sum (WRS) test.

- Increase the approved Type I decision error when using Scenario A or the Type II error when using Scenario B (MARSAME and the new MARSSIM revision include more on Scenario B).

- Change the documents prescribing the legally enforceable institutional controls [applies only to license termination under restricted conditions (10 CFR 20.1403)].

- Change the financial assurance method(s) required for license termination under restricted conditions [applies only to license termination under restricted conditions (10 CFR 20.1403)].

- Change the alternative criteria approved by the Commission under 10 CFR 20.1404 (applies only to license termination using alternate criteria).

1.5 References

1-1. NUREG-1575, “Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)”, Revision 1.

1-2. NRC Reg Guide 1.179, “Standard Format and Content of License Termination Plans for Nuclear Power Reactors,” Revision 2.

1-3. NUREG-1700, “Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans,” Revision 2.

1-4. NUREG-1757, “NRC Consolidated Decommissioning Guidance,” Volume 2, Revision 2.

2 SITE CHARACTERIZATION

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

Regulatory Guide 1.179 (Reference 1-2) 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 (Reference 1-4). Note that characterization is performed for both radiological and non-radiological contaminants. The radiological data provides important input to Chapters 5 and 6 of an LTP. 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 but as discussed above it is preferred that it be submitted 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.

MARSSIM states the following objectives for the different elements of the site release process:

The primary objectives of the Historical Site Assessment are to:

- identify potential sources of contamination
- determine whether or not sites pose a threat to human health and the environment
- differentiate impacted from non-impacted areas
- provide input to scoping and characterization survey designs
- provide an assessment of the likelihood of contaminant migration
- identify additional potential radiation sites related to the site being investigated

The primary objectives of a scoping survey are to:

- perform a preliminary hazard assessment
- support classification of all or part of the site as a Class 3 area
- evaluate whether the survey plan can be optimized for use in the characterization or final status surveys
- provide input to the characterization survey design if necessary

The primary objectives of a characterization survey are to:

- determine the nature and extent of the contamination

- collect data to support evaluation of remedial alternatives and technologies
- evaluate whether the survey plan can be optimized for use in the final status survey
- provide input to the final status survey design

Remedial action support surveys are conducted to:

- support remediation activities
- determine when a site or survey unit is ready for the final status survey
- provide updated estimates of site-specific parameters used for planning the final status survey

The primary objectives of the final status survey are to:

- select/verify survey unit classification
- demonstrate that the potential dose or risk from residual contamination is below the release criterion for each survey unit
- demonstrate that the potential dose or risk from small areas of elevated activity is below the release criterion for each survey unit

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.

It is recommended that site characterization be performed by an individual qualified and experienced in the process. Often the radiological and non-radiological characterizations are two separate and distinct areas of expertise.

As stated in NUREG 1.179 (Reference 1-2), "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 final status survey (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.

If continuing characterization is planned due to the inaccessibility of some site areas, the characterization section of the LTP should discuss how that data will be incorporated into the site decommissioning planning process as those areas become available for characterization. If the initial evaluation is conservative compared to data collected by the continuing characterization, it can be justified that no change is necessary to the final status survey plans for that area, otherwise it should be communicated to the NRC and a possible LTP revision may be needed.

The primary objectives of site characterization are:

- Provide data needs identified from the Historical Site Analysis
- Provide initial site assessment for survey area classification
- Provide data to determine ROCs, Mixture Fractions and variations/boundaries for which this data applies (this should consist of samples of significant contamination found across the site)
- Provide site data sufficient to support site dose modeling being performed (e.g., K_ds, groundwater gradient/flow, etc.)
- Determine activity in reference areas/materials, if needed
- Support 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 DandD process. If the likelihood of DRPs being present is low, then additional DRP surveys may be avoided.

If needed, the detection of DRPs may 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.

Section 3.2 further describes steps that can be taken to minimize the creation of DRP including the radiation control surveys needed to address control of DRPs.

Section 4.5 describes how the presence of DRPs should be addressed as a part of remediation activities and how isolation and control measures can minimize the spread of DRPs.

Section 5.1.3 describes how licensees have recently addressed, in their LTPs, how DRPs affect their Final Status Surveys.

Appendix H provides guidance on DRPs.

Additionally, site characterization is needed to (additional discussion of these topics is provided at end of this chapter):

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

2.1.1 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. Conversely, 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.

Four characterization types have been identified 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.

2.1.2 Types of Concrete Characterization

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 (3-to-4-inch diameter) affixed with a cutting surface on the concrete side. 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. Additionally, the process of creating the core can cause cross contamination of some portions of the core by contamination located in other portions of the core. This possibility needs to be considered in evaluating the analytical results for the core.

Concrete Dust Collection through Small Hollow-Core Drilling

This technique has emerged in the decommissioning industry in the past 10 years or so. It involves advancing a small (1/2 inch to 1-inch diameter) hollow bit that pulverizes the concrete (or other materials) as it advances. The concrete dust is drawn up into the hollow bit center to a collection system or a waste system. When collected, the dust is captured onto a filter paper and saved for later analysis. This type of system can be generally operated by one operator and when properly operated the precise collection depth is known. This system can also collect concrete samples or penetrate tens of meters in depth (as long as rebar is not encountered) and can operate from the lower activity side of a wall to the higher contaminated end to minimize personnel radiological hazards.

One potential disadvantage to this system is that the collection volumes and mass is generally much smaller than traditional coring. With such a constraint, the MDCs associated with the radiological analysis need to be well understood.

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 collect a small sample from a surface and transport the sampled material along a tube 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 is 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 minutes with a throughput rate of less than 10 minutes. 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, with the results indicating that the gamma activity for Am-241, Cs-137

& 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.

***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.

The use of in-situ gamma spectroscopy in place of scanning has been more widely used in decommissioning FSS as experience has grown. However, recognition of the following limitations should be considered. It is necessary to provide a good model of the contamination profile in the material being analyzed, so some characterization may be required. The material being surveyed should be relatively homogeneous. Characterization may need to assess the percent moisture or similar parameters to correct the measurement data for proper comparison.

In-Situ Gamma Spectroscopy should not be considered where past remediation for DRPs or expectation for potential DRPs exist. Additionally, as the activity indicated in a land area survey unit approaches the DCGL, complementary scanning and sampling should be considered. Examples of in-situ gamma spectroscopy being used in place of conventional scanning are given in Appendix A.

In-situ gamma spectroscopy has also been used to complement scanning building surfaces with beta sensitive instruments. As the two instruments have sensitivity for different radionuclides (beta versus gamma) care should be taken to use the correct input parameters when calibrating in-situ gamma spectroscopy devices for use on building surveys.

2.1.3 Numbers of Concrete Measurements/Samples

There is no discrete number of measurements or samples that can be prescribed for an effective characterization campaign. However, one should consider the statistical analysis of the data that will be needed as part of the data assessment, and this may differ based on the sampling/measurement methods used to collect the data.

For example, if sampling is intended to characterize the contaminant depth profile and there is an expectation that the radionuclide fractions may statistically change in different rooms/areas/buildings, then many samples (hundreds) may be needed to identify these characteristics since each unique area may only be represented by a much smaller number of samples. Therefore, consider what minimum sample size might apply to be able to provide an effective statistical analysis of each sample set.

For some prior nuclear power plant characterization campaigns, over 500 samples have been collected and analyzed by onsite gamma spectroscopy for depth profiling, and of these, 40 to 60 samples have been analyzed for beta, gamma, and alpha emitting radionuclides from various site locations. When identifying samples for offsite laboratory analysis, consider using samples with the highest overall activity since these will optimize the likelihood that the hard-to-detect (HTD) radionuclides will be identified and well quantified.

2.1.4 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. In many cases, the level of residual radioactivity in surface soils has been low and the collection of many samples may not be warranted since information regarding radionuclide fractions of both gamma emitting and HTD radionuclides is typically inconclusive.

Experience has shown that sample collection and sample preparation generally constitute the two largest sources of variation in soil sample results. The collection of the samples should be performed in a consistent manner (i.e., depth and size of collections area) and should consider sample preparation and the analysis of HTD radionuclides. Before analysis of the samples:

- Large stones, sticks, and other matter (such as vegetation) which is not representative of the exposure pathway should be removed from the sample.

- Samples should be homogenized and dried (except in cases where the samples are to be analyzed for tritium)

FSS sample procedures should be explicit in these regards and are subject to NRC review as part of the inspection process.

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 (which may have been subject to disturbance) 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.

2.1.5 Subsurface Soil Samples

Determining the locations for the collection of subsurface soil samples (depths greater than 15 cm and up to several meters below grade level per MARSSIM) is primarily dependent on:

- The potential for current or past subsurface leaks of radioactive material
- The potential for surface spills that include radionuclides that migrate readily through the surface soil to the subsurface.
- Whether there is a preferential pathway for surface contamination to penetrate to greater depths, such as along building foundations, drain tiles, well casings and/or artificially disturbed soils.

Likely sources are systems that carried liquid radioactive materials or adjacent to structures where contaminated materials were handled. (EPRI soil & groundwater remediation GLs. Reference 5.7)

Therefore, a soil sampling campaign should consider the potential for these bulleted pathways during the Historical Site Assessment (HSA), scoping, and characterization surveys.

2.2 Terminology Used in Radiological Analyses

It is important to use the terminology and methodology used in MARLAP (Reference 2-1) when discussing the conduct of radiological analyses/surveys and in interpreting the results of these. The following definitions are contained in MARLAP. More information on the determination and use of these terms is contained in MARLAP and MARSSIM (Reference 1-1).

2.2.1 Radiological Survey/Analyses Definition

- **Critical Value (Sc):** In the context of analyte detection, the minimum measured value (e.g., of the instrument signal or the analyte concentration) required to give confidence that a positive (nonzero) amount of analyte is present in the material analyzed. The critical value is sometimes called the critical level or decision level.
- **Minimum Detectable Amount (MDA):** The minimum detectable value of the amount of analyte in a sample. Same definition as the minimum detectable concentration but related to the quantity (activity) of a radionuclide rather than the concentration of a radionuclide. May be

called the "minimum detectable activity" when used to mean the activity of a radionuclide (see ANSI N13.30 and N42.23).

- **Minimum Detectable Concentration (MDC):** The minimum detectable value of the analyte concentration in a sample. ISO refers to the MDC as the minimum detectable value of the net state variable. They define this as the smallest (true) value of the net state variable that gives a specified probability that the value of the response variable will exceed its critical value-i.e., that the material analyzed is not blank.

2.2.2 MARLAP Recommendations Concerning Radiological Analyses

The following are some of the key recommendations in MARLAP concerning the use of the terms in the previous section:

- When an analyte detection decision is required, it should be made by comparing the gross signal, net signal, or measured analyte concentration to its corresponding critical value.
- The laboratory should choose expressions for the critical value and minimum detectable value that are appropriate for the structure and statistics of the measurement process. The client may specify the desired Type I and Type II error rates (both 5 % by default) but should not require particular equations for the critical value or the minimum detectable value without detailed knowledge of the measurement process.
- The laboratory should consider all sources of variance in the instrument signal (or other response variable) when calculating the critical value and minimum detectable value.
- The minimum detectable value (MDC or MDA) should be used only as a performance characteristic of the measurement process. It represents a hypothetical value of an analytical process which can be measured 95% of the time when the critical value is used as a detection criteria.
- A measurement result should never be compared to the minimum detectable value to make a detection decision.
- The laboratory should report each measurement result and its uncertainty as obtained even if the result is less than zero. The laboratory should never report a result as "less than MDC."
- The minimum detectable value should not be used for projects where the issue is quantification of the analyte and not detection. For these projects, MARLAP recommends the minimum quantifiable value as a more relevant performance characteristic of the measurement process. MARLAP neither encourages nor discourages the reporting of sample-specific MDCs with measurement results, so long as the recommendations stated above are followed.
- In the absence of true-positive sample results, analytical bias, or systematic errors from the analytical processes (cross contamination, inappropriate laboratory blanks, non-radioactive interferences etc.), approximately 5% (using the default k-alpha value) of all results would be expected to exceed the critical level when no activity above background is present. These are sometimes referred to a “false positives”. In cases where such results are suspect, the licensee should statistically evaluate the data to estimate the rate of these exceedances since they may

not represent true-positive results. If the rate of these exceedances is substantially greater than the expected false positive rate, then an investigation should be performed to understand its cause. Such an investigation should include consultation with the laboratory. Results that range between the critical level and the MDA may fall into this category and be compared to the laboratory-reported uncertainty.

2.3 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 2-4) and NUREG/CR-4289 (Reference 2-5).

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 2-2) 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.

2.3.1 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 DandD due to decay of the short-lived radionuclides.

- 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 DandD samples will likely differ in these characteristics.

- Mathematical Efficiency Models - If this approach is used for DandD 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.

- On-site sample analysis during the operation of the plant may not include drying and homogenization of soil samples. As discussed above, all the sample preparation protocols discussed above for surface soil samples need to be carried out on-site if the characterization samples are to be analyzed in the on-site laboratory.

- If the on-site laboratory is to be used for FSS samples, all the analysis and QA elements discussed below for off-site sample analysis, need to carried out by the on-site laboratory. For this reason, many sites have found it more cost effective to use an off-site laboratory for FSS samples.

2.3.2 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

The advantage of using an offsite commercial laboratory is generally related to improved data quality and specificity as well as ensuring that all samples submitted for the purpose of the detection of "hard-to-detect" radionuclides can also be subject to gamma spectroscopy within the same QA program. However, the analysis costs for this service can exceed that of the onsite laboratory and this should be carefully evaluated. In some characterization campaigns, data quality issues have been identified after expending substantial resources for onsite analysis whereas choosing an offsite laboratory from the beginning may have resulted in equivalent cost. A best practice to consider would be to participate in a laboratory cross check program to ensure validity of sample results.

2.4 Characterization of Groundwater Contamination

The characterization of the radionuclide contamination in groundwater is important to the determination of the dose to a future user of the site from existing and future groundwater contamination. To support the design of an adequate characterization of groundwater contamination for a site, there must be an understanding of the migration of radionuclides in groundwater.

Per NUREG 1757 (Reference 1-4), the use of a Conceptual Site Model (CSM) for a site is important for both the characterization of groundwater contamination and the abstraction and development of hydrological inputs for the dose models. A conceptual site model provides a hypothetical framework for contaminant source, geologic, hydrologic (including water usage), chemical, biologic, and demographic characteristics for the site. The CSM provides the basis for understanding flow and transport at the site for abstraction into a dose assessment model and is the starting point for numerical models if contamination is present in the surface water and/or groundwater. In general, CSMs should be updated as new information becomes available. The complexity of CSMs should be commensurate with site risk and at an appropriate level to demonstrate radiological criteria for license termination can be met. ASTM E1689-95 (2014) provides information on the development of CSMs that may be useful to licensees.

2.5 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 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.

Control of analytical data in terms of how it is received, controlled, verified, and then released for use can be beneficial to the decommissioning project. Aspects of this control that can be used to minimize the impact on the analytical results and provide consistency of those results include the following:

- Reviewing lab receipt reports to ensure that the samples were received as inspected (integrity, temperature, preservative, etc…)
- Verifying that the results are in terms of dry basis and that samples have been homogenized so that results in pCi/g are relevant and reliable.
- Verifying that water samples were or were NOT filtered prior to analysis, as specified by the licensee's analysis requirements
- Verifying that samples were processed as expected (screened for particle size, rocks and organic material removed, etc…)
- Adequate sample volumes were received
- A review of the data for statistical outliers, with a review to include the data point or to eliminate it. The decision should be documented**.**

Licensees can refer to the DQO and DQA sections of MARSSIM and MARLAP for further guidance on this topic.

2.5.1 Identifying Data Trends and Statistical Observations

The first step in the analysis process should be to organize the data by measurement quantity/sample matrix. For example, results for activity per unit mass (pCi/g) should be separated from activity per unit volume (pCi/L) or activity per unit area (pCi/cm^2, dpm/100 cm^2). Once the data is organized in this fashion, the data should be evaluated for trends. These trends could include observations by building, elevation, area, sample depth, and/or sample matrix as variation can occur even within the same building. The purpose of this trending is to ascertain which data sets can be grouped together in the event there are ROCs specific to such groupings. If such trending is identified, then the determination of activity fractions and ROCs may need to be kept separate for the analysis. Also, in evaluating the potential data trends, the reported measurement uncertainties should be considered for whether data should be included within a trend group.

As a final step in the process, the analytical data should be decay-corrected from the analysis reference date to the expected date of license termination. Note that for Am-241, this calculation differs from the other radionuclides since it is a progeny from the decay of Pu-241 and therefore requires an ingrowth calculation as follows. (Note that this need to normalize the data to a common date, such as at the projected License Termination date differs from the calculation of DCGLs where the computer codes used include the progeny in their derivation. Additionally, if the date is near-term, this ingrowth is not expected to be significant.)

$$(At)_2 = \left(\frac{\lambda_2 A_1^0}{\lambda_2 - \lambda_1}\right)\left(e^{-\lambda_1 t} - e^{-\lambda_2 t}\right) + A_2 e^{-\lambda_2 t}$$

Equation 2-1

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

2.5.2 Determining Radionuclide Activity Fractions

Radionuclide activity fractions are needed to;

- Evaluate data to determine the media that can be grouped based on similar fractions.
- Calculate the dose from insignificant contributors (ICs) and finalize the list of Radionuclides of Concern (ROCs).

It should be noted these calculations must use data with substantial activity to ensure statistical power. Soil and groundwater samples typically have low activity.

One of three methods (or any combination 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, $f_R A_{i,k,.75}$ using Equation 2-5.

$$f_R A_{i,k,.75} = \frac{R_{i,Cs-137,k,.75}}{\sum(i) R_{i,Cs-137,k,.75}}$$ **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 removes the activity weighting and gives 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

2.5.3 Determining Insignificant Radionuclides

NUREG-1757, Vol. 2, Rev. 2 (Reference 1-4) 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.

It should be noted that the determination of which radionuclides are insignificant should be made for each media for which DCGLs will be determined (i.e., water, soil). This may be difficult as the activity level in water and soil is typically very low at power plant sites.

Concerning the "insignificant radionuclides," as stated in NUREG-1757, Rev 2:

> *In general, the NRC does not require post-remediation sampling of the insignificant radionuclides, due to their low risk-significance. However, if there is a valid concern that the dose contributions of the postulated insignificant radionuclides could be significant following remediation, licensees may choose to manage this uncertainty as part of the DQO process (e.g., through post-remediation sampling of the insignificant radionuclides, similar to the approach used for surrogate radionuclides discussed in MARSSIM Section 4.3.2).*

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 Section 2.5.2 and Equation 2-6.

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

2.5.4 Final List of Radionuclides of Concern

Some DandD 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 analysis shows, only six 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 Dose Fraction	Soil 1.0 m 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

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

2.5.5 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 as an example.

The ratio of HTDs to gamma emitters is required to develop a surrogate relationship as defined in MARSSIM. The concentration of HTDs can 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. 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 this purpose.

Table 2-3: Sr-90 and Cs-137 Decayed Reported Concentrations, MDCs and Ratios

Sample ID	Sr-90 Decayed Reported Concentrations (pCi/g)	Sr-90 Decayed MDCs (pCi/g)	Cs-137 Decayed Reported Concentrations (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

From the values in Table 2-4, the 95th percentile value, or 0.197 in this case, could be chosen as a conservative initial value for the surrogate ratio in the subsequent FSS. A similar approach can be used for C-14. If use of this conservative initial value is determined to be impractical for use in the FSS, another value from the table would be used along with the required justification. The use of the conservative 95th percentile value may be justified for cases where the data is highly variable.

Developing a surrogate relationship between two radionuclides may be difficult as the concentrations of hard to detect radionuclides in soil and water are typically very low. The following are trends that have been observed and nuclear power plants (NPPs) concerning surrogates:

- Very few NPPs have needed to develop multiple surrogate relationships
- Surrogates can vary substantially even within a survey area. Therefore, combining many samples and establishing a conservative surrogate relationship is appropriate
- Positive analytes detected in groundwater have included H-3, Ni-63, Sr-90 and Cs-137. Surrogates generally have not been used in groundwater monitoring

The following are guidelines that could be followed considering the limitations that the sample data may present:

- Surrogate radionuclides presume some similarity in movement/causality is present. This may not hold up if the radionuclides are of significantly different chemical properties (e.g., non-soluble vs soluble in groundwater). If there is a significantly different chemical property anticipated amongst the radionuclides, which may be the case for certain soluble radionuclides such as H-3, Sr-90, Cs-137, then separate chemical analytical analysis for these radionuclides may be necessary as opposed to assuming a surrogate relationship exists.
- The primary and surrogate radionuclides should have a well-defined relationship. Statistically, a R value greater than 0.7 is typically considered a well correlated relationship.
- Surrogate ratios are for a given survey area and may not apply across the whole site. If a surrogate ratio is derived from one survey area for example, from a sample drain line, it may not be extrapolated to the whole site. It is the burden of the survey planner to prove that the radionuclide ratios used to develop the surrogate approach are representative for the area that the surrogates are being used.
- Extending the use of surrogates beyond one inferred radionuclide is difficult to do. This difficulty arises from demonstrating that a consistent ratio exists between two radionuclides and adding more creates greater complexity so more sampling and analysis will be required for justification of multiple surrogates.

2.6 Other Use of Site Characterization Data

The information in this is taken from the EPRI FSS Experience Report (Reference 2-3).

The 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 10CFR 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 the course of the 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.

2.6.1 Initial Survey Area Classification

After sufficient characterization information had 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.

2.6.2 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 then decommissioning progresses.

If characterization surveys are 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.

2.7 References

2-1 NUREG-1576, Multi-Agency Radiological Laboratory Analytical Protocols Manual (MARLAP), July 2004.

2-2 EPRI Report #3002000409, “EPRI Alpha Monitoring and Control Guidelines for Operating Nuclear Power Stations, Revision 2,” August 2013.

2-3 EPRI Report # 1015500, “Final Status Survey Experience Report,” March 2008.

2-4 U.S. Nuclear Regulatory Commission, NUREG/CR-3474, “Long-Lived Activation Products in Reactor Materials,” Pacific Northwest Laboratory, 1984.

2-5 U.S. Nuclear Regulatory Commission, NUREG/CR-4289, “Residual Radionuclide Concentration Within and Around Commercial Nuclear Power Plants; Origin, Distribution, Inventory, and Decommissioning Assessment,” Pacific Northwest Laboratory, 1985.

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 DandD regarding the radiological impacts of decommissioning activities; to ensure appropriate administrative and engineering controls are in place throughout the DandD 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 (Reference 1-2) 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 (DandD, engineering, waste, industrial health, regulatory affairs, etc.) to develop the required details of Chapter 3.

In the following sections, the focus should be on 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. Per NUREG-1700, an LTP should include a summary of all radiological control procedures already authorized under the existing license in addition to any changes or modifications.

As previously mentioned, the presence of discrete radioactive particles (DRPs) outside of contaminated areas has been a challenge to past decommissioning projects. 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 DandD process. A licensee should have in place radiation control procedures that:

- Define a DRP and the different levels of postings that will be used for access control.
- Describe the measures for conducting contamination control surveys and how to address DRPs if discovered.
- Provide for radiological surveys to be performed that are adequate to detect DRPs in survey units that could have DRPs based on site history, remediation activities, or proximity to areas where DRPs have been previously detected.
- Provide for timely and prudent actions in response to DRPs; direct that information related to DRPs be documented in the corrective action program, including the radionuclide, activity level, size, other qualitative details, etc.

If the likelihood of DRPs being present is low, then additional DRP surveys may be avoided.

If needed, the detection of DRPs may 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.

3.3 Structures at License Termination

Provide a table of site structures denoting which site structures will be removed or will remain in place 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 existing radiological control procedures are adequate as discussed earlier in this chapter. Chapter 4 discusses soil and groundwater remediation techniques.

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 3-1).

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 U.S. 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 Received 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 approval for disposal of decommissioning waste at such a facility through this process.

3.5.3 U.S. NRC Waste Exemption Process

The U.S. 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 NRC 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 NRC 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 involved in the requests. 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 3-2).

3.5.4 Other Radioactive Waste Considerations

Prior to demolition, building surfaces needed to be characterized for paint containing polychlorinated biphenyls (PCBs) and other hazardous constituents. Paint containing concentrations equal to or greater than 50 mg/kg (50 ppm) of PCBs or other hazardous constitutes such as lead based paint needs to be remediated and the waste may require disposal as mixed waste. PCB contamination of soil has also been a major chemical and mixed waste issue at some decommissioning sites.

Mastic coatings containing asbestos may be present on some subsurface foundations. These coatings will need to be remediated if special dispensation can be obtained from the appropriate regulatory authority.

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.

3.7 References

3-1 EPRI Report #1024844, “Basis for National and International Low Activity and Very Low-Level Waste Disposal Classifications,” March 2012.

3-2 EPRI Report #3002000587, “Generic Technical Basis for Implementing a Very Low-Level Waste Category for Disposal of Low Activity Radioactive Wastes,” December 2013

4 REMEDIATION PLANS

4.1 Introduction and Background

Per the guidance OF NUREG-1700 (Reference 1-3), this chapter of the LTP needs to include a summary and discussion on how a facility and its 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 (Reference 4-1).

Chapter 4 includes a discussion of the remediation methods and techniques that the licensee will use to perform the remediation of site areas to meet the NRC criteria for license termination. As discussed in the introduction for Chapter 3, Chapter 4 also requires input from site experts performing DandD 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, (Reference 1-4) 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. In addition, the ALARA evaluations concerning remediation discussed below, licensees should be mindful that it is important that remediation plans must also consider occupational exposure monitoring and environmental monitoring plans. The radiological environmental monitoring program may need to be adjusted and, perhaps, the Offsite Dose Calculation Manual may need to be modified to quantify new effluent pathways and other changes such as:

- Gaseous releases from buildings that no longer exhaust through the original plant stack
- Liquid releases to a different body of water than that used during plant operation
- Reduction in dilution flow used during liquid releases
- Changes to the equipment/techniques used to monitor releases and the calibration procedures for these

Information in this chapter is taken from the EPRI FSS Experience Report (Reference 2-3) unless otherwise noted.

4.2 Lessons Learned

The following are examples of some of the more important lessons learned in site remediation:

- Post remediation ALARA evaluations have generally not resulted in the need for additional remediation work, as site remediation efforts have generally resulted in sufficiently low residual radioactivity levels.

- Remediation of excavations in areas where the groundwater table is near to the surface has proven to be very challenging. Sites have needed to employ methods such as grout and freeze walls, sheet piling and/or high flow rate area dewatering.

- Remediation of concrete has been shown to require more than shallow surface removal due to contamination at depth and in cracks and crevices in the concrete at some plant sites.

4.3 Remediation Levels and ALARA Evaluations

When dismantlement and decontamination actions are completed, residual radioactivity may remain on building surfaces, on-site soils and groundwater. Residual radioactivity must satisfy the ALARA provisions of 10 CFR 20, Subpart E. To satisfy this provision, it must be shown that further remediation is not required to reduce the levels of contamination beyond that which are present to meet the ALARA requirement when a survey unit passes its Final Status Survey.

NUREG-1757, Volume 2, Revision 2 provides guidance on two options that can be used to demonstrate ALARA compliance for decommissioning sites. The first option is a performance based ALARA compliance, which can be found in NUREG-1757, Volume 2, Section 6.3.6, “Compliance Methods at the Time of Decommissioning.” This performance-based ALARA compliance method is likely an extension of ALARA committee activities that were occurring during the operation of the facility.

The second option described in NUREG-1757, Volume 2, Revision 2 is the predetermined compliance measure, which is necessary for a proposed restricted release scenario, this is not the case for an unrestricted release scenario. Note that both options are briefly discussed in section 6.3.6 of NUREG-1757, Volume 2, Appendix N. The NRC guidance for the predetermined compliance approach from the NUREG is summarized in the following:

The following few paragraphs are taken from Appendix N of NUREG-1757, Volume 2, Revision 2, which provides NRC guidance on performing ALARA evaluations during decommissioning using the predetermined compliance approach discussed above. This information provides an overview of Appendix N and the reader can refer to the NUREG for additional details on methods that can be used to perform ALARA evaluations.

> *To terminate a license, a licensee must demonstrate that it has met the dose criteria in Subpart E and the requirements for reducing exposures as low as is reasonably achievable (ALARA). This section describes methods acceptable to the NRC staff for determining when it is reasonable to further reduce the (future) exposures of members of the public below the dose criteria. This section does not apply to, nor replace guidance for, operational ALARA programs.*

This guidance does involve the same principle as the operational ALARA guidance, as described in NRC Regulatory Guide 8.8, Revision 3, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low as Is Reasonably Achievable," issued June 1978: "Reasonably achievable" is judged by considering the state of technology and the economics of improvements in relation to all the benefits from these improvements.

For ALARA, as it relates to the license termination criteria, all licensees should use typical good practices, such as floor and wall washing, removal of readily removable radioactivity in buildings or in soil areas, and other good housekeeping practices. In addition, in the Final Status Survey Report (FSSR), licensees should describe how they employed these practices to achieve the final activity levels.

In light of the conservatism in the building surface and surface soil generic screening levels developed by the NRC, the staff presumes, absent information to the contrary, that licensees who remediate building surfaces or soil to the generic screening levels [given in Appendix H of NUREG 1757 Volume 2, Revision 2] *do not need to provide analyses to demonstrate that these screening levels are ALARA. In addition, if residual radioactivity cannot be detected, it may be assumed that it has been reduced to levels that are ALARA. An ALARA analysis is also unnecessary in cases where soil removal is performed. Therefore, the licensee may not need to conduct an explicit analysis to meet the ALARA requirement.*

In general, a method for determining whether exposures that would result from a proposed license termination approach are ALARA would have the following characteristics.

- *The method is simple. The method for most licensee applications should be simple, because the effort needed for very sophisticated models cannot generally be justified. In an ALARA analysis of a remediation action, the primary benefit (i.e., the collective radiation dose that may actually be averted in the future) is uncertain because future land uses, the number of people that may actually occupy a site, and the types of exposure scenarios are all uncertain. These uncertainties mean that the future collective dose cannot be known with precision. Because of the inherent limitation on the ability to precisely determine the future collective dose at a particular site, it is not useful to perform a complex analysis when a simple analysis can be appropriate. Licensees may use more complex or site-specific analyses if more appropriate for their specific situations (e.g., restricted release analyses, situations that include a number of unquantifiable benefits and costs).*

- *The method is not biased and uses appropriate dose modeling to relate concentrations to dose. The determination of ALARA should not be biased. This is different from demonstrating compliance with a dose limit. The analyses for dose assessments and surveys for compliance with the dose criteria described in this volume include a reasonably conservative bias for demonstrating compliance. Unlike a demonstration of compliance, an ALARA analysis is an optimization technique that seeks the proper balance between costs and benefits below the dose limit. To achieve a proper balance, each factor in the ALARA analysis should be determined with as little bias as possible. If the ALARA analysis were intentionally biased, it would likely cause a misallocation of resources and could deprive society of the benefits from other uses of the resources. Thus, the ALARA analysis should provide an unbiased analysis of the remediation action,* *which can both avert future dose (a benefit to society) and incur costs (a potential detriment, because it can deprive future generations of the return on the investment of this money). Sections N.2 through N.6* [of NUREG 1757] *discuss the methods that licensees should use in estimating benefits and detriments, or costs, including scenarios, models, and parameters for relating concentration to dose at a site.*

- *The method is usable as a planning tool for remediation. Before starting a remediation action, the licensee should be able to determine generally what concentration of residual radioactivity would require a remediation action to meet the ALARA requirement. It would be inefficient if the licensee could not tell whether the area would pass the ALARA test until after the remediation. Establishing ALARA post remediation could also result in it being less likely for a licensee to remediate below the dose limit(s) because of the additional manpower startup costs associated with performing additional remediation.*

- *As much as possible, the method uses the results of surveys conducted for other purposes. The demonstration that the ALARA requirement has been met should not require surveys beyond those already performed for other purposes, such as the characterization survey and the FSS. It would be inefficient (and unnecessary) to collect additional sets of measurements to demonstrate that remediation actions were taken wherever appropriate to meet the ALARA requirement, if the licensee could use measurements undertaken for other purposes.*

ALARA as it Applies to NRC Decommissioning Regulations

ALARA, as defined in 10 CFR 20.1003, means:

making every reasonable effort to maintain exposures to radiation as far below the dose limits in this part as is practical consistent with the purpose for which the licensed activity is undertaken, taking into account the state of technology, the economics of improvements in relation to state of technology, the economics of improvements in relation to benefits to the public health and safety, and other societal and socioeconomic considerations, and in relation to utilization of nuclear energy and licensed materials in the public interest.

The general requirement for ALARA is stated in 10 CFR 20.1101(b)

The licensee shall use, to the extent practical, procedures and engineering controls based upon sound radiation protection principles to achieve occupational doses and doses to members of the public that are ALARA.

For license termination with either restricted or unrestricted release, doses to a member of the public must meet the release criteria (e.g., 0.25 mSv [25 mrem] per year for unrestricted release or restricted release with institutional controls in place) but also ALARA. Therefore, ALARA should be considered in conjunction with the applicable release criteria including unrestricted release, restricted release with institutional controls in effect, and restricted release with institutional controls no longer in effect.

The licensee's implementation of the general ALARA principle from 10 CFR 20.1101(b) should be appropriate for the specific regulatory basis being utilized, which could include 10 CFR 20.1402 ("Radiological Criteria for Unrestricted Use"), 10 CFR 20.1403 ("Criteria for License Termination

Under Restricted Conditions"), or 10 CFR 20.1404 (Alternate Criteria for License Termination) for unrestricted release, restricted release, or alternative release criteria, respectively.

An 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 1-4). Documentation of ALARA evaluations must be included in the Final Status Survey report for each survey area. It should be noted that ALARA evaluations need to utilize the most recent values for costs and benefits such as:

- The most recent regulator provided dollar value of a rem of dose averted
- Remediation costs need to adjusted for inflation
- Most recent cost of capital if equipment is to be purchased

There may be situations where the site is required by a regulatory authority or stakeholder agreement to remediate to lower dose criteria or concentrations than those required by the NRC, but these more restrictive numbers should not be included in the LTP. A separate document of agreement should be used to describe these criteria and/or values. Experiences with these type of more restrictive criteria are given in Appendix D. When an FSS is performed at a site that falls under lower site released criteria than that required by the NRC, the licensees have typically used operational DCGL that are below the NRC approved DCGLs to address these more restrictive state requirements.

4.3.1 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 Appendix F 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.

Additionally, as discussed in Chapter 5 and 6 of this report, the dose from pathways related to groundwater must be included with that from other media in showing compliance with the site release criteria. As is needed for other media in compliance equations such the one shown in Appendix B, an ALARA cost-benefit analysis needs to be performed when there is detectable contamination in groundwater. The next section summarizes the type of techniques that can be used to perform remediation.

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 which is discussed below in Chapters 5 and 6. 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
- Bedrock
- Nonstructural plant systems (including interior surfaces of process piping and components)
- Groundwater

4.4.1 Structures

Methods for remediating metal structures may include a variety of techniques ranging from water washing to surface material removal. Several factors determine the choice of the remediation method for a given area, including: the size of the contaminated area, the extent of contamination, surface material, depth of contamination, and accessibility. Remediation activities for an area may include wiping, vacuuming, and washing with low- or high-pressure applications.

Concrete surfaces on contaminated structures can be remediated to levels meeting the radiological criteria defined in a site's LTP. The debris materials are disposed of at a licensed radioactive material disposal facility.

4.4.2 Shallow Remediation Techniques

Techniques that can be placed in the category of shallow or localized methods are:

- Concrete shavers
- Media blasting systems
- Small hand-held tools for localized remediation (i.e., scalers and grinders)

4.4.3 Aggressive Remediation Techniques

As can be seen in the last section, the "shallow" remediation techniques generally remove a relatively thin layer of concrete. These devices may take repeated passes or additional time at a location to remove contamination located deep below the surface or in cracks or construction joints in the concrete. During some decommissioning projects, the use of "aggressive" removal techniques has been found be to the most cost-effective approach. "Aggressive" in this context refers to the ability of this type of equipment to remediate larger and deeper amounts of concrete faster than the "shallow" techniques. This section will describe some of the most common approaches used when removal of a greater depth of radioactive contamination is desired.

In performing aggressive concrete remediation, larger, more powerful equipment is utilized. The size of the remediation equipment that can be used is defined by the space available in the remediation area. For example:

- If a building is to be remediated in a standing condition, the size of the remediation equipment is restricted to the size of access openings that exist or can be economically created in the remediation location.
- If a large quantity of remediation is required and the work area is large, it most likely is cost justified to create a large opening in the building so that large equipment can access the inside of the structure.
- If complete removal of a structure is to be accomplished, work is generally conducted from outside of the building using large equipment.

Several examples of these situations will be described later in this chapter following a brief description of some commonly used equipment or techniques.

Single Operator Jackhammers

These devices are commonly used and referred to as industrial pneumatic jackhammers. They can remove a significant depth of concrete and are faster than the smaller devices of this type described above. They are labor intensive compared to the techniques described below and are more difficult to safely operate than some of the other techniques. They could be the method of choice in hard to access areas.

Large Hydraulic Hammers

Large hydraulic hammers, in the context of nuclear decommissioning, are standard industrial excavators with hydraulic hammers mounted as end effectors. This configuration of the equipment is often called a "Hoe-Ram." There are several sizes of these machines with larger models being more effective for the wholesale removal of buildings or concrete inside large structures, i.e., such as inside the containment buildings at power plants. For a building with a large garage door and a reasonably high ceiling, one of the smaller sized excavators can pass into the building to perform the remediation. Use of this equipment is generally the quickest way to perform concrete remediation when large areas are to be remediated.

One difficulty with the use of this type of equipment inside of buildings or tents is that they are diesel-powered and create carbon monoxide and other hazardous exhaust. When used inside a restrictive enclosure, adequate ventilation needs to be provided for the safety of the workers, or the exhaust must be directed outside of the enclosure.

For demolition from outside of a structure, the use of this equipment is straightforward. As significant concrete dust can be created during demolition, spraying, or misting with water is required for dust suppression.

Diamond Wire Cutting

Cutting with a diamond wire has been successfully used during several decommissioning projects. It is particularly effective when the concrete is very thick and when complete removal of the structure is planned. With diamond wire cutting, an approximately 0.4 inch (1 cm) thick wire assembly, with industrial grade diamonds held by the wire completely along its length, is drawn across the concrete surface. The drive system for the wire keeps the tension on the wire steady as it passes through the concrete structure. After the system is set up, it is operated remotely as the wires can break and present a safety hazard from the loose ends until the system is shutdown. Maintenance is needed in the form of repairing broken wires and replacing diamonds when they become worn.

An advantage of this technique is the preparation of the cut pieces for shipment. For concrete that has a low enough activity level, the large concrete pieces created can be placed in a large polymer bag that is qualified as a shipping package and transported to a radioactive waste disposal site without further preparation.

4.4.4 Soils

When site characterization shows that the site has subsurface soil contamination that may require remediation, the input parameters used to calculate soil DCGLs must reflect the area, depth and thickness profile of the subsurface contamination. This ensures that the potential dose from subsurface contamination, including that from the groundwater pathway, is properly considered in the determination of soil DCGLs.

Soils not meeting the applicable DCGLs need to be removed and disposed of as radioactive waste. Licensees may develop different DCGLs for surface soils and subsurface soils. Additionally, a licensee may excavate soil that exceeds a volumetric bases DCGL and then perform an FSS on the surface of the open excavation. In this case, small areas that exceed the DCGLw(s) may pass an elevated measurement comparison using $DCGL_{EMC}$(s) and therefore pass the FSS in the same manner as would be appropriate for a surface soil FSS.

Offsite fill or on-site material shown to meet the site’s DCGLs can be used to replace the excavated materials. The ongoing site characterization process establishes the location, depth, and extent of soil contamination. As needed, additional investigations are performed to ensure that any soil contamination profiles that may change during the remediation actions are adequately identified and characterized.

Some state regulators prohibit the use of certain media resulting from remediation activities as backfill of excavations on the site. The prohibition of the use of concrete demolition debris as backfill at Maine Yankee during their decommissioning is an example of this. Licensees should negotiate the site restoration requirements with the state prior to submitting the LTP so the end state of the site can be determined, and dose modeling is performed consistent with the end state/site restoration.

Removal of soil is typically performed using excavators. If the site has a relatively shallow groundwater table, the remediation area may need to be dewatered prior to soil removal to maintain the sidewalls of the excavation. If the removed soil is stockpiled prior to packaging for disposal, the potential impact of this stockpile on groundwater should be evaluated. Experience with backfilling of excavations is discussed in Chapter 5.

Bedrock

In some cases, bedrock underlying the soil has become contaminated. If the contamination has entered the fractures of the bedrock, blasting may be needed to break up the bedrock prior to removal by excavators.

If a thin layer of soil continues to need remediation, a large vacuum truck is a method that has been used.

4.4.5 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").

4.4.6 Groundwater

A site may have detectable contamination in groundwater that is determined to require remediation to meet the site release criteria or other criteria such as the EPA/NRC MOU (Reference 4-2). It should be noted that the release criteria given in the EPA/NRC MOU is more stringent than that which meets NRC regulations. These differences are discussed in Appendix F.

The source of this contamination can be from building basements, soil, bedrock and the possible new impact of demolition and remediation activities (including the draining of contaminated soils). Although the remediation techniques for these media are discussed in the sections above, there are some additional techniques that may be appropriate as discussed in the following.

The EPRI Groundwater and Soil Remediation Guidelines (Reference 4-3) provides the types of information necessary to evaluate remediation options with respect to technical feasibility, safety, and cost. While the process described in this guideline does not prescribe a particular remediation option, it does describe a number of options for consideration at a nuclear power plant site with soil and/or groundwater contamination and the interaction between the two. More importantly, the guideline provides an outline of a decision-making process for the remediation of soil and groundwater that can support a well-defined business decision, meet regulatory requirements (including an ALARA evaluation), and maintain the health and safety of the public and the environment.

Once a site has determined the need for remediation, remediation strategies will need to be evaluated and implemented. The remediation strategies may include passive remediation (i.e., monitored natural attenuation) or aggressive remediation (e.g., removal of soils, pumping of groundwater, etc.) techniques. Factors such as operational and radiological safety, effectiveness of the remediation, and cost of remediation should be included in the evaluation of each remediation strategy. Viable remediation options that could be used to achieve the site release limits for a contaminated area will likely vary depending on the hydrogeological conditions and radionuclide concentrations present in that area. The EPRI Groundwater and Soil Remediation Guidelines discuss the various remediation options and how they may apply to various site-specific situations. Information is included on topics such as:

The effect of the following on remediation evaluations including ALARA cost benefit evaluations:

- Effectiveness and the ability to meet site release limits
- Radwaste costs

Remediation options discussed including the potential effectiveness of each:

- Monitored Natural Attenuation
- Augmented Monitored Natural Attenuation
- Pump and Discharge
- Pump and Treat
- Engineered Confinement
- Permeable Reactive Barriers
- Soil and Bedrock Removal
- Soil Mixing
  - Called Intentional Mixing by the NRC in this context is when soil that exceeds site release limits is allowed, after NRC approval, in certain cases which are defined in NRC Regulatory Issue Summary 2004-08 (Reference 4-4)
  - Evaluation of this option should consider the connection of pre-remediation subsurface soil contamination to the potential for causing and/or increasing groundwater contamination, when:
    - having to drain soils or
    - mixing contaminated and clean subsurface soils to reduce overall concentrations to meet soil DCGLs
- Biological Remediation Options
- Combinations of the Above Options

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.

Although limited activities may occur within a previously surveyed area prior to license termination, care should be taken not to utilize it for storage or handling of impacted materials that originated outside of the survey unit, even if the materials were previously surveyed for release.

The spread of Discrete Radioactive Particles (DRPs) has been challenging at some decommissioning sites. Isolation and control procedures need to address the surveys needed including:

- Those needed to ensure that DRPs are not present in an area where remediation has been completed
- Resurvey of areas where isolation and control has been implemented but where there is suspicion that DRPs may have been introduced

Section 3.2 discussed other radiation control procedures that address DRPs.

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

4.6 References

4-1 Code of Federal Regulations, Title 10, Part 20.1402, "Radiological Criteria for Unrestricted Use."

4-2 U.S. Environmental Protection Agency and U.S. Nuclear Regulatory Commission, Memorandum of Understanding on Consultation and Finality on Decommissioning and Decontamination of Contaminated Sites, October 2002.

4-3 EPRI Report #1023464, “Groundwater and Soil Remediation Guidelines for Nuclear Power Plants Public Edition,” July 2011.

4-4 NRC Regulatory Issue Summary 2004-08, NRC Regulatory Issue Summary 2004-08, “Results of the License Termination Rule Analysis,” May 2004

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 2-3).

The regulations applicable to this area of review are 10 CFR 20.1402, 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 Survey (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 (DandD 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 the U.S. Nuclear Regulatory Commission Part 50 or Part 72 license).

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

An FSS Plan description is required as Chapter 5 of the site's License Termination Plan (LTP) and therefore is a license condition that <u>must</u> be followed.

The objective of this document is to provide a high-level survey summary of the FSS process, and emphasize 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 based on contamination potential. Survey units are established to facilitate the FSS process and the statistical analysis of the data. It should be noted that MARSSIM provides guidance for the design of surveys of structures and land areas but not for commodities such as subsurface structures and buried piping. Licensees have typically utilized alternate methodologies to calculate the dose from commodities such as these. These alternate methodologies may utilize DCGLs specific to subsurface structures and buried piping that are developed for the site. An example of these types of DCGLs is given in Appendix F, Section F.6.3. When buried pipe DCGLs have been developed, they are used along with the contamination levels on the buried piping in a survey unit to determine the dose from the buried piping. This dose is then added to the dose from other media in the survey unit, such as soil, subsurface soil and groundwater to show compliance with the site release criteria (i.e., 25 mrem/yr). Alternately, individual dose calculations can be performed for individual sections of buried piping.

For an area that includes a structure, there may be multiple components included in the compliance equation such as subsurface structures, subsurface soil, embedded piping and groundwater. As with a land area, DCGLs may be developed or alternate dose calculations may be performed for each media. The individual doses from each applicable media are then added in a compliance equation to show that the site release criteria (i.e., 25 mrem/yr) is met. Appendix A provides additional information about the use of compliance equations.

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 have 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. Definitions for the various classifications are given in Section 2.6.

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)	<100 m^2
Land areas	<2,000 m^2
Class 2:	
Structures (floor area)	100 m^2 < area < 1,000 m^2
Land areas	2,000 m^2 < area < 10,000 m^2
Class 3:	
Structures (floor area)	no limit
Land areas	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). As these DCGLs correspond to a dose of 25 mrem/yr, they are typically called “Base Case” DCGLs. A detailed discussion of the different types of DCGLs is provided later in this chapter and in Appendix B. However, a typical commercial reactor site has multiple source terms (types of contaminated media) – including:

- surface soil
- subsurface soil
- surface and subsurface structures
- buried and embedded pipe
- existing groundwater contamination

A detailed discussion of the development of DCGLs, Pathway Dose Conversion Factors for use assessing groundwater contamination, and other methods of converting measurement results into projected dose to a user of the site after license termination is included in Section 6 and Appendix F of this document. 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 other dose modeling codes.

As the potential dose from all applicable pathways must be included in showing compliance with the site release criteria, the base case DCGLs (or dose conversion factors) 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. These reduced DCGLs are typically called Operational DCGLs. An example of this reduction process is included in Appendix B.

The methods for applying these DCGLs and dose conversion factors in Final Status Surveys and other assessments of potential post license termination dose are discussed in the remainder of this chapter.

The FSS process involves:

- survey design
- survey implementation
- data assessment

It should be noted that while the NRC approves methods for the FSS, there are assumptions associated with an FSS plan of which the licensee should be aware. If actual circumstances vary from the assumptions, a modification to the methods may be warranted and the NRC will accept the application of more conservative and suitable methods. For example, most scanning is established for a minimum area of diffuse contamination (e.g., 0.25 m^2). If smaller areas or discrete materials are present, then scanning procedures may need to be modified to be more sensitive to the contaminating material. These variations should be documented in the Final Status Survey Report along with how the data quality objectives (DQOs) and measurement quality objectives (MQOs) were adjusted.

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 1-1), 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 MARSSIM Data Life Cycle process embellishes the DQO process and its use should be a commitment of licensees. The MARSSIM Data Life Cycle should be followed during various phases of the MARSSIM process including the characterization, survey process implementation and data assessment phases. The Oak Ridge Associated Universities (ORAU) Professional Training Program (PTP) has created a “MARSSIM Cheat Sheet” that summarizes the MARSSIM process and how the DQO and Data Life Cycles processes factor into it. This cheat sheet is included as Appendix J.

MARSSIM lists the following 17 objectives of the DQO process that apply during the site release survey process:

For the design of the Historical Site Assessment

1. Identify potential sources of contamination
2. Determine whether or not sites pose a threat to human health and the environment
3. Differentiate impacted from non-impacted areas
4. Provide input to scoping and characterization survey designs
5. Provide an assessment of the likelihood of contaminant migration
6. Identify additional potential radiation sites related to the site being investigated

For the design of the Scoping Survey

1. Perform a preliminary hazard assessment

2. Support classification of all or part of the site as a Class 3 area
3. Evaluate whether survey plan can be optimized for use in characterization or final status survey
4. Provide input to the characterization survey design

For the design of the Characterization Survey

1. Determine the nature and amount of contamination
2. Evaluate remedial alternatives and technologies
3. Evaluate whether survey plan can be optimized for use in final status survey
4. Provide input to the final status survey design

For the design of the Final Status Survey

1. Select/verify survey unit classification
2. Demonstrate that potential dose or risk from residual contamination is below the release criterion for each survey unit

Demonstrate that potential dose or risk from residual elevated areas is below the release criterion for each survey unit. 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 radiological release criterion of the NRC License Termination Rule (Reference 4-1).

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. Definitions and discussion on making detection evaluations and ensuring adequate instrument sensitivity (i.e., MDCs) are given in Section 2.2 and later in this chapter***.*** 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 1-1, 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 at which 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. These are also often called Base Case DCGLs.

$DCGL_{EMC}$ - $DCGL_{EMC}$ 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.

Licensees have also used specific names for other types of DCGL such as $DCGL_{BP}$ for use in the survey of buried pipe. It should be noted that the base case DCGLs are often corrected for insignificant contributors and then smaller values corresponding to lower doses are allocated amongst the various media. The media specific DCGLs, such as $DCGL_{BP}$, are often used to guide the remediation efforts and design the FSS. Although the 25 mrem/y $DCGL_W$ values as approved in the LTP are unchanged, these lower “operational” DCGLs are often used to demonstrate compliance for the various media.

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 1-1) 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 5-1), 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
- Appropriate classification of survey area

5.1.4 Survey Considerations for Suspected Discrete Radioactive Particle Areas

Survey methods described in the License Termination Plan (LTP) should include consideration of areas that are suspected to contain discrete radioactive particles (DRPs) and/or discrete radioactive objects. One acceptable approach is to develop a site-specific procedure(s) to describe the appropriate methods for identifying and controlling DRPs. The following is a high-level summary of the site-specific DRP Control procedure(s) that has been acceptable to the NRC.

In this regard:

- Section 3.2 describes steps that can be taken to minimize the creation of DRP including the radiation control surveys needed to address control of DRPs.
- Section 4.5 describes how the presence of DRPs should be addressed as a part of remediation activities and how isolation and control measures can minimize the spread of DRPs.

The following describes how Fort Calhoun has recently addressed, in their approved LTP, how DRPs affect their Final Status Surveys.

A site-specific procedure concerning the conduct of Final Status Surveys can been developed that:

- Defines a DRP and/or discrete radioactive objects

- Provides that radiological surveys be performed that are adequate to detect DRPs in survey units that could have DRPs based on site history, remediation activities, or proximity to areas where DRPs have been previously detected. The surveys that are conducted as part of the site's Final Status Survey should be consistent with guidance in the MARSSIM DQO process to ensure that the survey results are of sufficient quality and quantity to support the final decision on allowing unrestricted use of the site.

If needed, the detection of DRPs may 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, as of the publication of this revision to NEI 22-01, 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 1-1).

Appendix H provides more information on addressing DRP during the Final Status Survey.

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. In FSS, scanning is conducted in an effort to identify areas of elevated activity. When found, additional survey beyond that which was determined during survey planning is needed to determine the areal extent so that the levels can be compared to the DCGLs for that building area. If surface contamination levels identified in these investigation surveys exceed the DCGLs, they can be used to perform an elevated measurement comparison. If the survey unit had been classified as MARSSIM Class 2, and elevated areas exceeding the DCGLs are found, the survey unit needs to be reclassified as MARSSIM Class 1 and the additional scanning appropriate for a Class 1 survey unit performed.

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.

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 1-4) 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.

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 per Revision 1 of MARSSIM (Reference 1-1). As can be seen in this table, when the potential for contamination is higher (i.e., in a Class 1 Area), higher scan coverage is required. Revision 2 of MARSSIM is expected to be issued in 2025 and contains a revision to scan coverage requirements. Table 5-4 shows the revised scan coverage requirements.

Table 5-3: Traditional Scanning Coverage Requirements (per MARSSIM Revision 1)

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

Table 5-4: Scanning Coverage Requirements (per MARSSIM Revision 2)

Area Classification	Scanning and Direct Measurements and/or Sampling Survey		Scan-Only Survey
	Scanning	Direct Measurements or Samples	Scanning
Class 1	100%	Number of data points from statistical test (See Sections 5.3.3 and 5.3.4 of MARSSIM); additional measurements may be necessary for small areas of elevated activity (see Section 5.3.5 of MARSSIM)	100%
Class 2	10-100% systematic and judgment “Scan Area” calculated by: $\frac{(10 - \Delta/\sigma)}{10} \times 100\%$	Number of data points from statistical test (See Sections 5.3.3 and 5.3.4 of MARSSIM);	10-100% systematic and judgment “Scan Area” 10% or as calculated by: $\frac{(10 - \Delta/\sigma)}{10} \times 100\%$ whichever is larger
Class 3	10-100% judgment “Scan AREA” calculated by: $\frac{(10 - \Delta/\sigma)}{10} \times 100\%$	Number of data points from statistical test (See Sections 5.3.3 and 5.3.4 of MARSSIM);	10-100% judgment “Scan Area” 10% or as calculated by: $\frac{(10 - \Delta/\sigma)}{10} \times 100\%$ whichever is larger

It should be noted that the revised scanning requirement in Revision 2 of MARSSIM has two major revisions as follows:

- Requirements for “Scan-Only” surveys – These are the types of surveys that are described in Section 5.2.3 along with examples provided in Appendices A and H.

- An equation is provided for use in calculating the required scan area for Class 2 and Class 3 areas. This equation included the factor "Δ/σ" which is the relative shift described below. Additionally, it should be noted that when the survey for a unit is to be “Scan- Only” for Class 2 and Class 3 areas, at least 10% of the unit’s survey area should be scanned.

The equation in Table 5-4 includes the factor ”Δ/σ” which is defined as the relative shift. The relative shift is equal to the width of the gray region “Δ” which is equal to the “$DCGL_W$ – LBGR” divided by an estimate of the uncertainty “σ” which can be calculated using characterization data for the survey unit. The “LBGR” in this equation is the Lower Bound of the Gray Region and is a site-specific variable generally chosen to be a conservative (slightly higher) estimate of the concentration of residual radioactive material remaining in the survey unit.

Additional Requirements for Scan-Only Surveys

MARSSIM Revision 2 provides additional guidance for Scan-Only surveys as follows:

In Section 4.8.3.1:

MARSSIM does not recommend "scan-only" FSSs. However, through the DQO process and consultation with the regulator, an FSS based on scanning measurements alone might be allowed. Items that should be kept in mind while investigating a scan-only survey are:

- *data and location logging*
- *reproducibility of the measurements (e.g., fixing a detector at a constant distance from a surface)*
- *MDCs*
- *scanning speed and operator training*
- *data integrity and security*
- *selecting an appropriately sized area for elevated measurement comparison calculations*

In Section 5.3.6.1:

…a scan-only approach should be used only for circumstances where the measurement method has sufficient detection capability to meet the MQOs to both quantify the average concentration of radioactive material in the survey unit and identify areas of elevated concentrations of radioactive material. To ensure that this is the case, the scan MDC (for the scan system) should be less than 50 percent of the $DCGL_W$. The scan-only methodology will require validation, which likely requires collecting some percentage of samples for laboratory analysis to compare with results from the same location. Other MQOs should be met as well, including the MQO for measurement method uncertainty at the $DCGL_W$.

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 and sensitivity so that augmenting the scanning survey with fixed measurements 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.

Appendix A gives some examples of the “advanced technologies.”

5.2.4 Gross Activity DCGLs

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

$$DCGL_{GA} = \frac{1}{\sum_{1}^{n} \frac{f_i}{DCGL_i}}$$

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 NUREG-1757 (Reference 1-4), 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 no 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 attributed to 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 2.5 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 illustrates 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 [i.e., >10% of their DCGL] as discussed in the last section) based on process knowledge, historical data, or characterization.

- Screen HTD radionuclides using the 10% rule described earlier in this section.

- Radionuclides not screened out will require a surrogate DCGL. Surrogate relationships can be determined from the samples results using one of the methods described below.

- Using the ratios developed by a process like that in Section 2.5, develop a surrogate DCGL for each HTD radionuclide, as in Equation 5-2:

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

  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

Including the contribution of Hard-to-Detect Radionuclides (HTD) to the dose from structure surfaces can be a complex task. Alpha radionuclides generally present the greatest challenge as they are not detected by beta sensitive instrumentation and normally have DCGLs that are approximately an order of magnitude below that of the primary beta/gamma-emitting radionuclides (Co-60 and Cs-137). In general, it is expected that separate alpha and beta surface activity measurements (beyond the required operational radiation protection surveys) will not be necessary at a plant with the typical beta to alpha ratios of 1000 or higher. When this is the case, surrogate measurements can instead be used for alpha surface activity assessments.

For building surfaces where the beta to alpha ratio is significantly lower, the use of the surrogate relationship will result in relatively low surrogate DCGLs for the beta survey, potentially to where the sensitivity requirements for the beta survey instruments would be very difficult if not impractical to achieve. In this situation, a separate alpha surface final status survey would need to be conducted. This would allow the beta surface survey to be conducted using standard survey equipment and methods.

In cases where alpha scan surveys are required, MDCs must be quantified differently than those for beta-gamma surveys because the background count rate from a typical alpha survey instrument is nearly zero (1 to 3 counts per minute typically). Since the time that an area of alpha activity is under the probe varies and the background count rates of alpha survey instruments are so low, it is not practical to determine a fixed MDC for scanning. Instead, it is more useful to determine the probability of detecting an area of contamination at a predetermined DCGL for given scan rates.

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

Condition of Surface to be Surveyed

Addressing contamination in cracks and construction joints can be a major challenge to the conduct of a surface FSS. As concrete contamination is generally due to fluid system leakage, cracks and joints are areas where higher contamination levels occur. The sensitivity and interpretation of surface survey instruments normally assumes a relatively smooth surface and does not include the effect of cracks.

Regulator verification surveys have included the use of gamma sensitive instruments like a 2” x 2” sodium iodine detector to assess contamination in cracks. To address this effect, licensees have needed to perform much more remediation in the locations of cracks and construction joints. This process of "chasing cracks" has led in some cases to the removal of the total thickness of the structure floor.

Another challenge of addressing the problem of contamination in cracks concerns the concrete inside of containment buildings in PWRs. The containment buildings at many plants were constructed with a steel liner on the inside of the primary building reinforced concrete structure. This liner was intended to be leak tight to preclude the escape of contamination in the event of an accident. These liners were very effective at containing fluid system leakage from spreading to the concrete outside the liner. This has allowed the release of the concrete outside the liner for conventional landfill disposal or reuse on-site as backfill.

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 helps to ensure that the collected data meet the needs of the final status survey. The collected data may be used as the reference area data set when using the WRS 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 5-5 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 5-5 are derived from both NUREG-1507 (Reference 5-1) and from experience at the Connecticut Yankee (CY) Plant.

Table 5-5: Typical Media Specific Backgrounds (Reference 5-1)

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 c 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 1.5-inch	3,000 - 6,000 cpm for soil
	2-inch by 2-inch	8,000 cpm - 16,000 cpm for soil

Depending on the values of the applicable DCGLs, an alternative method to using material specific backgrounds has been accepted for use by the NRC during final status surveys. This alternative method involves the determination of the ambient area background in the survey unit and will only be applicable to beta-gamma detecting instruments. This determination is made prior to performing a final status survey at location(s) within a survey area that is of sufficient distance (or attenuation) from the surfaces to eliminate beta particles originating from the surfaces from reaching the detector. It should be noted that holding the detector a sufficient distance away from the surface to eliminate betas may be inadequate for Sr-90/Y-90 (10 ft beta in air) and that background measurements should (either the sufficient distance or beta absorber method) be collected in areas away from suspected contamination.

An alternative to a "sufficient distance" is to place a beta absorber over the detector of sufficient thickness to fully attenuate the maximum energy beta particles from the ROCs. This process should be performed in various locations within a survey unit to ensure consistency in the background level with a statistical evaluation to demonstrate the absence of elevated background areas.

At such location(s), or attenuation, the average ambient background radiation is due only to ambient gamma radiation and will be a background component of all surface measurements. The average background determined at these locations can be used as a conservative estimate since it is expected to be less than the material-specific background for the material in the room as it does not fully account for the naturally occurring radioactivity in the materials. Using this lower ambient background will result in conservative calculated residual radioactivity levels. If the average background reading exceeds a predetermined expected value for the area, the survey would be terminated, and an investigation performed to determine and eliminate the reason for the elevated reading. This average background reading would be subtracted from each of the survey unit readings and the Sign Test would be applied.

Whether or not they are radionuclide specific, all background measurements should account for both spatial variability over the area being assessed, and the precision of the instrument or method being used to make the measurements. Thus, the same materials or areas may require more than one background assessment to provide the requisite background information for the various survey instruments or methods expected to be used for final status surveys. The result of these background assessments provides the basis for determining the mean and its associated standard deviation. Additionally, apparent biases can be very noticeable in final reported data if non-conservatively determining the ambient background/reference material concentrations. If most or all net results are negative, this is indicative of a non-conservative bias being present and effort should be made to better determine a suitable background. If no contamination is truly present, the net instrument results should fluctuate around zero with some positive and some negative. The average of the readings may be slightly negative but should be "zeroed" for demonstrating compliance with the dose criterion.

Another method for considering background would be to avoid any background subtraction and assume all surface measurements represent the presence of residual radioactivity. This approach would overestimate the level of residual radioactivity and its implementation would not be dependent on background fluctuations. This method can be considered in cases where the background is a very small fraction of the DCGLs where background subtraction would not significantly impact the survey assessment results.

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 5-1).

5.2.8 Building FSS Techniques and Alternate Approaches

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

Alternate Approaches to Final Status Survey

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

For characterization of dose from backfill, backfill samples either actual values and/or critical values should be used for characterization of the backfill material. For dose estimation purposes a sum-of-fractions of actual values for each sample can be used and if the average of the sum-of-fractions for all the samples taken is negative, the dose may be zeroed. When doing sum-of-fractions for individual samples, results less than critical level may be zeroed and remaining sample results that are greater than critical values used to determine dose.

Subsurface Structures

The calculation performed to determine dose using a Subsurface Structure Dose Model (formerly called 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 SSDM are provided in Chapter 6.

It should be noted that "Basement Fill Model" is an industry term and the methodology used in the model has varied from site to site. When considering the use of this type of model to determine the dose from basement structures and/or materials used to fill these basements, this variation in the experiences needs to be considered. Additionally, due to this variation, the use of a Basement Fill Model approach should not be considered pre-approved as it will be reviewed in detail by the NRC.

In this document, for the cases where licensees have used the term "Basement Fill Model" in the past, the model will be called a Subsurface Structure Dose Model (SSDM) with the parenthetical: ("called Basement Fill Model in the LTP").

Use of Soil and Demolition Debris as Backfill

The following few sub-sections describe experiences at some of the plants that have used soil and/or concrete demolition debris, that has met the site release criteria, as backfill of excavations or basements.

Multiple options exist for backfill sources, such as offsite areas, nonimpacted areas on site, or impacted areas on site that have been appropriately surveyed. Both NUREG-1757 Volume 2 Revision 2, "Consolidated Decommissioning Guidance" (Reference 1-4) and Interim Staff Guidance DUWP-ISG-02, "Radiological Survey and Dose Modeling of the Subsurface to Support License Termination" (Reference 5-6), contain guidance related to the use or reuse of soil and demolition debris for backfill of excavations and subgrade structures.

Backfill reused from onsite areas should be characterized to the rigor of an FSS. Backfill soil or demolition debris concentrations are compared to the appropriate DCGL to ensure that the risk is not underestimated. In the case of soil used for backfill, it is appropriate to use scans and sampling while being excavated or immediately after the soil is excavated and placed in a laydown area. For the reuse of demolition debris, the ability to detect and quantify volumetric contamination is particularly important.

If a licensee is to assume there is no added residual radioactivity in the backfill, support should be provided for this assumption. If there is uncertainty that backfill soils are non-impacted, NUREG-1757 references the potential use of a statistical test such as a Scenario B type analysis to show indistinguishability from background.

For soil or demolition debris that contains licensed material, appropriate scanning and sampling is performed in order to determine the risk, or dose, impact as part of the compliance dose modeling and calculation. In one of the examples for the Zion site that follow in this section, a license conservatively assumed that soil contained licensed material at the minimum detectable sensitivity of the survey instrumentation even though there were no detectable radionuclides present. Although conservative assumptions are sometimes used to simplify dose modeling, the use of minimum detectable concentration values as real characterization data is not required or recommended.

Licensees should continue to discuss proposed soil and demolition debris reuse plans with the NRC, as there are potentially complex issues associated with radiological measurement capabilities and site-specific dose assessments. In addition, discussion with other stakeholders regarding potential radiological or non-radiological impacts related to the reuse of soil or demolition debris as backfill at a decommissioning site may be warranted.

<u>Yankee Rowe Plant</u>

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 beds 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.

<u>Zion Plant – Use of soil as backfill</u>

The experiences from the Zion plant in this chapter are from the Zion LTP (Reference 5-2). 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). DUWP-ISG-02 has guidance that can be cited for “small decision units.”

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

- 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 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 consisted 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 were analyzed by gamma spectroscopy.

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 a Subsurface Structure Dose Model (SSDM called BFM in their LTP) at Zion is first needed.

The SSDM 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 doses 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 was 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

As can be seen from the earlier discussion, the conduct of an FSS for a building is a complex and therefore expensive task. Buildings that have housed no contaminated systems normally exhibit little or no contamination. For these types of buildings, a “free release” survey has been used during many power plant decommissioning projects. In these “free release” surveys, a percentage of the building surfaces are surveyed to unconditional release limits (i.e., no detectable plant-related radioactive material). Additionally, as these structures are expected to have very little or no contamination, they can be surveyed as MARSSIM Class 3 survey areas as described elsewhere in this chapter. For further information on “free release” refer to the Multi-Agency Radiation Survey and Assessment of Materials and Equipment Manual (MARSAME) NUREG 1575, Supplement 1.

5.2.10 Survey Protocol for Non-Structural Systems and Components

The guidance provided in MARSSIM for conducting final status surveys does not include guidance for conducting final status surveys for non-structural systems or components.

In general, non-structural components and systems (excluding buried and embedded piping) have been surveyed to the “free release” criteria normally used to release materials during the operation of the plant. The NRC guidance in MARSAME (Reference 5-5) can also be used in designing these types of surveys.

Embedded pipe represents medium- to large-bore pipe sleeve penetrations (up to 42 inch or 1m) or small-bore piping (4 inch to 12 inch [0.1 m to 0.3 m]) that were built into concrete walls and run through structures including walls, ceilings, and floors. At most plants, the length of the piping for each segment was generally short, approximately the length of the thickness of the structure that the pipe penetrated. In most cases the piping penetrated straight into the wall or floor of the room. The total length of this type of pipe at Connecticut Yankee was estimated to be less than 1000 feet (300 m), segregated into a substantial number of individual segments as would be typical for a nuclear power plant.

The development of release limits for embedded piping is discussed in Section 6.2.2 and Appendix F***.***

The limits that were approved for the CY site for this type of pipe sleeves were the Building Occupancy DCGLs discussed in Section 6. As with other building surface surveys, options included:

- Adjustment of Gross Beta/Gamma Limits for HTDNs such as alpha and pure beta radionuclides
- Separate alpha surface surveys performed when beta to alpha ratios are relatively low

Additional discussion of limits used to release embedded and buried piping for Connecticut Yankee is given in Section 6.

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)]. 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 criteria 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 NaI detector operated by survey technicians. The technician walks slowly across the survey unit while swinging the detector slowly back and forth. If an increase in meter response occurs, 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.

It should be noted the “meter responses” discussed in the last paragraph are audible meter responses. Surveyors should be aware and trained to ensure that scanning being performed is consistent with, or more conservative than that approved in the LTP. If discrete radioactive particles have been detected, adjustments to the scanning process may be needed to ensure an adequate sensitivity.

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 the 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 Appendices A and H.

Advanced Technology

Although most plants have conducted Final Status Surveys 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 $DCGL_W$.

In cases where these coverage requirements cannot be achieved by an advanced survey technology or where the MDC is too large relative to the applicable $DCGL_W$, 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 $DCGL_W$.(see examples in appendix A)

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.

It should be noted that in the case of in-situ gamma spectroscopy the “viewing area” for each in-situ measurement needs to be large enough, with sufficient sensitivity, such that any “missed” areas in the survey unit would be smaller than the largest allowed under the elevated measurement criteria. An example of how this was achieved at the Rancho Seco site is explained in Appendix A.

- 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.

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 (which is present at a lesser concentration). Although there are typically other radionuclides present in the natural background (e.g., uranium, thorium, radium, etc.), these are not Radionuclides of Concern (ROC) at nuclear power plants.

Determining the background concentration involves identifying a suitable area (generally off-site in undisturbed soils) 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 gain NRC approval in their LTP (as well as discussing these with other stakeholders) for subtraction of background values 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

MARSSIM provides clear guidance on the conduct of a Final Status Survey for surface soil in land areas. Additionally, recent NRC Interim Staff Guidance in DUWP-ISG-02, "Radiological Survey and Dose Modeling of the Subsurface to Support License Termination," (Reference 5-6) provides guidance on Final Status Surveys for land areas that have subsurface contamination. As the Connecticut Yankee site exhibited subsurface contamination, the experience at this plant with addressing this issue is presented in the following subsections.

It should be noted as noted that for excavations resulting from remediation activities, an FSS of the excavation should be performed prior to backfilling. Additional surveys of backfilled materials may also be warranted. Surveying backfilled/impacted areas prior to covering them with non-impacted materials/grading could avoid additional costs for subsurface sampling. Communicate with regulators when surveying/sampling of an excavation is an industrial safety concern and the use of alternative surveying/sampling methods are desired.

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.

It should be noted that this approach used at CY was based on the subsurface characterization at that site and may not be appropriate for other sites. Depending on the depth and thickness of residual radioactivity, depth discrete sampling may be needed. Surface dose pathways may dominate, and elevated surface residual radioactivity should not be composited with subsurface residual radioactivity if there is vertical heterogeneity and a potential to underestimate the dose due to the compositing method. In general, the sampling approach should be compatible with the dose modeling assumptions used to derive DCGLs. Likewise, the dose modeling should reflect the actual distribution of residual radioactivity at the site and consider lateral and vertical heterogeneity in contaminant distributions as appropriate.

Surface deposited radioactivity generally follows a logarithmic depth profile. That is, if the surface soils pass FSS, then the deeper soils will generally pass as well, provided that the exposure scenario includes an assumption of contamination >15 cm (>6 in.). This only becomes a problem when the contamination was not surface deposited or there is a preferential pathway for contamination to migrate at depth. Both are special cases. During FSS, licensees should ensure that no layer of clean soil is covering and obscuring a lens of contaminated soil. This is accomplished through characterization and careful checks at depth during Remedial Action Support Surveys when spots are found that need further remediation. Documenting this assures closure of the survey unit.

Homogenization of subsurface soils over an entire 1 m (3 ft) of depth will yield an average contaminant concentration over the whole 1 m (3 ft). This information is only useful if the results are compared to the DCGLs from an exposure scenario that considers such a large layer of contaminated subsurface soil (and technical support is provided showing that the dose results are not sensitive to the depth and thickness of contamination thereby allowing the DCGLs to be derived over a relatively large thickness). This method also obscures any information about the depth profile of the contamination, in a manner that may be non-conservative. For example, the concentration of contaminant in the 16 cm to 31 cm (6 in to 12 in) below ground surface layer may be significantly more important than the concentration in the 85 cm to 100 cm (33 in to 39 in) below ground surface layer.

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.

It should be noted that if the radionuclide ratios are to be determined using final status survey data, MARSSIM Section 4.3.2 recommends that at least 10% of the measurements (both direct measurements and samples) include analyses for all radionuclides of concern including HTD radionuclides.

<u>Subsurface Sample Density</u>

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 was classified as Class B or C due to the lower potential for subsurface contamination.

Table 5-6: 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 (by drilling through the floor and collecting a sample below the floor) 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 5-2). 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.

It should be noted that no regulatory guidance existed in the early 2000s when the sampling density given in Table 5-6 was defined for CY. Because the exposure scenarios, dominant pathways, radionuclides of concern, and risk may differ for surface and subsurface residual radioactivity, the classification, survey unit size, DCGLs, Area Factors and investigation levels will likely differ for surface versus subsurface residual radioactivity. The recently published NRC Interim Staff Guidance in DUWP-ISG-02, "Radiological Survey and Dose Modeling of the Subsurface to Support License Termination" (Reference 5-6) provides additional guidance on consideration of exposure pathways and application of MARSSIM to subsurface residual radioactivity.

Zion Subsurface Soil FSS

Subsurface soil as defined at Zion, 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) was remediated. The remediation process included performing a Remedial Action Support Survey (RASS) 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 at Zion was 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 was high confidence, based on site characterization results, that deeper samples did not result in higher concentrations.

Alternatively, if the contamination consists of gamma radionuclides, a NaI detector or intrinsic germanium detector of sufficient sensitivity to detect residual radioactivity at small fraction of 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. This approach could be useful when the excavation is unsafe for access by survey personnel. If the detector identifies the presence of contamination at a significant fraction of the Operational DCGL (e.g., >25% of the Operational DCGL), additional steps would be required to allow the performance of confirmatory investigation and analyses of soil samples of the suspect areas.

If significant quantities of HTD nuclides have been detected in the subsurface at a particular area, sampling would be required to assess the potential dose from these radionuclides.

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, 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.

It should be noted that NRC revised the relevant guidance in Appendix G to NUREG-1757, Revision 2, Section G.3.1as follows:

> *Because the MARSSIM FSS method was designed specifically for residual radioactivity in surface soils, if significant amounts of residual radioactivity are located at depth (e.g., significant quantities of residual radioactivity in soils deeper than approximately 15 centimeters), the presence of subsurface residual radioactivity should be taken into consideration in designing the FSS. The licensee should first determine whether it needs surveys of subsurface residual*

radioactivity. The HSA and other surveys will play an important role in determining whether there is likely to be residual radioactivity in the subsurface. Modeling can also be used to supplement survey data to determine the potential for residual radioactivity to be present in significant quantities in subsurface soils or groundwater due to environmental transport. If the survey data and supplemental modeling indicate that there is little likelihood of significant subsurface residual radioactivity, then subsurface surveys are likely unnecessary.

In Class 1 open land survey units, a subsurface soil sample was 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.

As was discussed above for CY, the approach used to analyze subsurface samples should be based on the subsurface characterization at the site. Depending on the depth and thickness of residual radioactivity, depth discrete sampling may be needed. Surface dose pathways may dominate, and elevated surface residual radioactivity should not be composited with subsurface residual radioactivity if there is vertical heterogeneity and a potential to underestimate the dose due to the compositing method. In general, the sampling approach should be compatible with the dose modeling assumptions used to derive DCGLs. Likewise, the dose modeling should reflect the actual distribution of residual radioactivity at the site and consider lateral and vertical heterogeneity in contaminant distributions as appropriate.

Homogenization of subsurface soils over an entire 1 m (3 ft) of depth will yield an average contaminant concentration over the whole 1 m (3 ft). This information is only useful if the results are compared to the DCGLs from an exposure scenario that considers such a large layer of contaminated subsurface soil (and technical support is provided showing that the dose results are not sensitive to the depth and thickness of contamination thereby allowing the DCGLs to be derived over a relatively large thickness).

All subsurface soil samples taken at Zion 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 at Zion remain 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 Zion site-specific Base Case DCGLs. 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 subjected to “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 of any sub slab soil samples taken were analyzed for the initial suite of HTD radionuclides as well as 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.

It should be noted that NRC staff has addressed subsurface characterization and survey issues in NUREG-1757, Volume 2, Appendix G. Additionally, the contamination concern map discussed in NUREG/CR-7021 (Reference 5-3) describes the extent, location, and significance of residual radioactivity relative to the decision criteria. This map can be developed with the aid of visualization, geographic information system and geostatistical software. Also, the guidance provided in NRC's Draft Technical Letter Report, "Guidance on Surveys for Subsurface Radiological Contaminants," (Reference 5-4) summarizes industry accepted practices and references for NRC-proposed activities including historic applications, all focused on subsurface soils.

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

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.

Licensees should provide an evaluation of groundwater monitoring network and a plan for the sampling program that meets the needs specified in the LTP for the final status survey. In compliance with the Groundwater Protection Initiative defined by NEI document, NEI 07-07, Rev. 1 (Reference 5-7), all US nuclear power plants sites (both operational and in decommissioning) have carried out a groundwater protection program that includes a plan and procedures for assessing the contamination in groundwater including assessing the adequacy of the groundwater monitoring well network. For most sites these programs have been active since 2007. These programs should continue after the plant is permanently shutdown and include an assessment of the adequacy of monitoring well network and sampling program to provide the data needs specified in the LTP for final status surveys. It should be noted that if a requirement stated in the LTP disagrees or exceeds what is stated in NEI 07-07, the LTP requirement supersedes what is in NEI 07-07.

Concerning remediation activities performed to address contamination in groundwater, NEI document, NEI 07-07, Rev. 1 (Reference 5-7) 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 4-3) provides detailed guidance as to how to meet the guidance statement of NEI 07-07 as well as some of the methods and approaches that can be used to remediate a site to address existing groundwater contamination.

For sites where elevated levels (i.e., those that could results in groundwater concentrations at approach or exceed the EPA MCLs) of mobile radionuclides (see Appendix K) are present that could be released soil and groundwater, the groundwater assessment plan should consider the need for trends in groundwater data over some period of time after completion of potential soil disturbance activities such as excavations, building demolition, or other demolition activities that may mobilize radionuclides. The period of time is site-dependent but should include two seasonal high precipitation periods (i.e., at least 18 months). While the evaluation and plan are most directly applicable to the choice to use the maximum concentration at the site for the compliance calculation, aspects would also be applicable to sites where the licensee has chosen to apply sophisticated tools to estimate different concentrations to different areas of the site.

Appendix G contains examples of characterization, remediation and final dose assessment of groundwater contamination at a number of sites.

The computer models used to calculate the DCGLs for soil (or dose conversion factors for other media) typically include the groundwater pathway dose from radionuclides that leach from the soil or other media over time. These models do not include dose from contamination that is already present in groundwater at the time of license termination (i.e., Existing Groundwater Dose). Section F.4 has a discussion on an approach that can be used to calculate the dose from radionuclide contamination already present in groundwater. Appendix F also provides examples of the dose impact of other buried commodities such as structures and buried piping.

5.3.5 Bedrock Assessments

Exposed bedrock from the demolition of structures or the remediation of contaminated soils creates another area requiring a methodical radiological assessment. Again, as Connecticut Yankee performed the greatest amount of remediation to bedrock, the CY experience is provided as the best example of addressing these types of assessments.

Several areas of the CY site were excavated to bedrock either through the demolition of buildings or the removal of contaminated soils. Excavation in the Tank Farm area to bedrock allowed for data to be assessed on the potential magnitude and distribution of contamination within the bedrock. This assessment includes the determination of:

- the degree of contamination on the bedrock surface,
- the degree of contamination migration into bedrock cracks and fissures, and
- the observation of surface conditions of the bedrock.

After gross removal of material covering the bedrock was completed, readily removable materials were removed from the bedrock surfaces using the “Vacuum Truck,” which carried a high flow rate vacuum equipped with a HEPA filter. Following remediation, the radiological conditions were assessed prior to backfilling the excavation. The backfill ultimately consisted of clean fill or fill shown to meet the soil DCGLs.

The dose pathway that applies to such open bedrock excavations is from potential future groundwater contamination since other pathways such as direct exposure, and plant uptake would not apply to this material (backfill provides substantial shielding to the surface and farming plants could not be grown in bedrock). Therefore, the post-remediation field assessment and dose assessment methods focus on the radioactivity inventory potentially available to future uncontaminated groundwater in contact with bedrock for each bedrock excavation.

The post remediation monitoring of the bedrock area was through the installation and sampling of groundwater monitoring wells. Once the bedrock area was backfilled, any dewatering wells and/or dewatering pumps, needed to suppress the groundwater were turned off allowing the groundwater to return to an equilibrium condition in the unconsolidated backfill materials. Monitoring wells were installed at locations to cover the plume migration paths determined by modeling of groundwater flow through unconsolidated material and bedrock. The installed monitoring wells were sampled quarterly for at least 18 months to include two high water table springtime periods when groundwater has its greatest impact.

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.

Additionally, if the site has a relatively shallow groundwater table, additional measures such as those described in the following section may be needed.

As discussed earlier, CY needed to perform a great deal of soil remediation to meet site release limits for groundwater. Much of the soil to be remediated was located under the groundwater table. Extensive dewatering prior to and during remediation was required to allow the soil to be removed and packaged in a dry state. The extracted groundwater contained high levels of minerals and total suspended solids (TSS) in addition to low levels of radionuclides. To meet the site National Pollutant Discharge Elimination System (NPDES) limits, TSS levels would need to be reduced. Utilizing the site radioactive liquid waste processing system for this activity would have been extremely expensive as processing capacity up to 450 gpm was necessary. The high levels of minerals and solids in the groundwater would have created very large volumes of radioactive spent filter cartridges and ion exchange media.

Connecticut Yankee worked with the Connecticut Department of Energy & Environmental Protection (CT DEEP) 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).

5.3.8 Other Considerations and Option for Surveying Subsurface Media

NRC Interim Staff Guidance in DUWP-ISG-02, “Radiological Survey and Dose Modeling of the Subsurface to Support License Termination,” (Reference 5-6) (Sections 2.6 and 2.7) provides guidance on additional considerations and options that can be used in the survey and/or sampling of subsurface soil and survey of subsurface structural surfaces and other commodities.

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.

Interpreting the results from a survey is the 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 5-7 can be used. See Section 5.3 for discussion of Background Reference Areas that is applicable to Table 5-7 and Table 5-8.

Table 5-7: Initial Evaluation of Final Status Survey Results (Background Reference Area Used) (Reference MARSSIM)

Evaluation Result	Conclusion
Difference between largest survey unit measurement and the smallest reference area measurement is less than the $DCGL_W$	Survey unit meets release criterion
Difference of the survey unit average and the reference area average is greater than the $DCGL_W$	Survey unit does not meet release criterion
Difference between any survey unit measurement and any reference area measurement greater than $DCGL_W$ and the difference of survey unit average and reference area average is less than the $DCGL_W$	Conduct WRS test and elevated measurement comparison

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 5-8 can be used.

Table 5-8: Initial Evaluation of Final Status Survey Results (Background Reference Area Not Used)

Evaluation Result	Conclusion
All measurements less than $DCGL_W$	Survey unit meets release criterion
Average greater than $DCGL_W$	Survey unit does not meet release criterion
Any measurement greater than $DCGL_W$ and the average less than $DCGL_W$	Conduct Sign test and elevated measurement comparison

The MARSSIM manual, in particular Sections 8.2 and 9, contains additional guidance on FSS data assessment.

5.5 References

5-1 NUREG-1507, "Minimum Detectable Concentrations with Typical Radiation Survey Instruments for Various Contaminants and Field Conditions," Revision 1 August 2020.

5-2 EnergySolutions, “Zion Station Restoration Project License Termination Plan,” Revision 2.

5-3 NUREG/CR-7021, “A Subsurface Decision Model for Supporting Environmental Compliance,” January 2012.

5-4 NRC *Draft Technical Letter Report: Guidance on Surveys for Subsurface Radiological Contaminants*, April 2021 (Adams Accession No. ML21123A229)

5-5 NUREG-1575, Supplement 1, “Multi-Agency Radiation Survey and Assessment of Materials and Equipment Manual (MARSAME)

5-6 NRC Interim Staff Guidance, DUWP-ISG-02, “Radiological Survey and Dose Modeling of the Subsurface to Support License Termination,” September 2024 (ML#24197A219)

5-7 Nuclear Energy Institute, NEI 07-07 Revision 1, “Industry Groundwater Protection Initiative – Final Guidance Document, Rev. 1,” March 2019.

6 COMPLIANCE WITH RADIOLOGICAL CRITERIA FOR LICENSE TERMINATION

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

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 dose-based requirements for release of a site are defined in 10 CFR Part 20, "Standards for Protection Against Radiation," Subpart E, "Radiological Criteria for License Termination," (aka, the License Termination Rule or LTR) (Reference 4-1). Guidance is provided in NUREG-1757, Volume 2, Revision 2, "Consolidated Decommissioning Guidance, Characterization, Survey and Determination of Radiological Criteria," (Reference 1-4) on acceptable methods for demonstrating that dose criteria are met through dose modeling to develop clean-up criteria (or DCGLs) and radiological surveys to demonstrate mean or median concentrations in the survey unit are less than the release criteria while minimizing decision errors.

NRC guidance in DUWP-ISG-02, "Radiological Survey and Dose Modeling of the Subsurface to Support License Termination," (Reference 5-6) provides guidance on acceptable methods to consider risk from existing groundwater contamination. 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 25 mrem/yr (0.25 mSv/yr). This approach has resulted in very low DCGLs which, in some cases, required extensive remediation to achieve.

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 in the following sections.

Resident Farmer Scenario

This scenario assumes that a family resides at the site after license termination. This family grows all of 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 DandD code or the RESidual RADioactivity (RESRAD) dose modeling computer code developed by Argonne National Laboratory have been used.

The DandD 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 DandD code is readily available on the Oak Ridge Associated Universities website at Reference 6-1, 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 6-2). Both codes can be downloaded and used free of charge from their respective websites.

Parameter Selection Using the Results of Sensitivity Analysis

NUREG-1757, Volume 2, Revision 2, provides acceptable methods for demonstrating that dose criteria are met through dose modeling to develop clean-up criteria (or DCGLs) and radiological survey to demonstrate mean or median concentrations in the survey unit are less than the release criteria while minimizing decision errors.

NUREG-1757, Volume 2, Revision 2 and DUWP-ISG-02 (Reference 5-6) provide guidance on:

- Using parameter distributions in sensitivity analyses *and*
- How to use the results of these sensitivity analyses to select input parameters for use in dose modeling.

The following four paragraphs summarize some of the key points of guidance in these NRC documents as stated in the NRC comments on revision 1 of NEI 22-01 provided in reference 6-12.

NUREG-1757, Volume 2, Revision 2, Appendix I indicates that for risk-significant parameters, additional support may be needed for deterministic values used in the compliance demonstration to ensure that the doses are not under-estimated (i.e., that the 25th or 75th percentile values may not be demonstrably conservative for broad parameter distributions such as distribution coefficients or K_ds) (See the next subsection for additional guidance on the determination of the K_d input parameter).

The overall recommendation of NUREG-1757, Volume 2, Revision 2, is to use site-specific values or otherwise provide support for the values selected commensurate with the risk-significance of the parameter. Appendices I and Q of NUREG 1757 do not state that probability density functions from the literature require increased justification and cannot be used in probabilistic sensitivity analysis; however, it is advisable to use site-specific information, if available, in both sensitivity analysis (e.g., K_d parameter distributions based on site-specific soil type and geochemistry) and for assessment of risk/DCGL development.

NUREG-1757, Volume 2, Revision 2, states that RESRAD defaults for physical parameters important to dose are not acceptable for use without further justification. RESRAD default parameter distributions can be used to perform sensitivity analysis to determine the importance of the parameter on dose. Additionally, default behavioral and metabolic parameters in NUREG/CR-5512, Volume 3, are acceptable for use without further justification (e.g., see Table I.11 and associated text in NUREG-1757, Volume 2, Rev. 2).

NRC has provided an update to the guidance on parameter distributions in NUREG/CR-6697, "Development of Probabilistic RESRAD 6.0 and RESRAD-BUILD 3.0 Computer Codes," which has been cited in several examples provided in this report. This updated guidance is provided in NUREG/CR 7267 "Default Parameter Values and Distributions in RESRAD-ONSITE V7.2, RESRAD-BUILD V3.5, and RESRAD-OFFSITE V4.0 Computer Codes." (Reference 6-13).

As noted in NUREG-1757, the licensee may use representative distributions or values for the physical parameters. The licensee is not required to routinely adopt worst-case, bounding, upper- or lower-percentile, or other overly conservative values in defining distributions.

Additional Guidance in Determining Input Values for the Distribution Coefficient (Kd) Parameter

NRC guidance in DUWP-ISG-02 (Reference 5-6) provides several methods and considerations for estimating dose modeling parameter input values for the distribution coefficients (Kds) for use in the determination of DCGLs for a site. Based on NRC's guidance in DUWP-ISG-02, measurements of sorption coefficients are not required at sites. The guidance further suggests that a graded approach should be selected based on site conditions, data availability, dose modeling approach, and treatment of Kd inputs in RESRAD (e.g., site-based uncertainty versus selection 25/75 percentile based on generic tables). In particular, the ISG states the following in Section 3.3.2, *Determining if a Parameter Value is Risk-Significant*:

> *Therefore, as a starting point, only Kd values for radionuclides that have a potential to lead to doses greater than 0.025 mSv/yr may require additional support, and only if they are found to be risk-significant. For example, if there is little uncertainty in the Kd value, additional support is likely unneeded.*

As is described in detail in Appendix K, experiences from power plant sites have shown that only:

- A few radionuclides have been detected in soil in quantities that would sum to a dose that would exceed 2.5 mrem/yr (0.025 mSv/yr)
- Only a very few radionuclides have been detected in groundwater
- Of these few radionuclides, only H-3 has shown a sensitivity where the DCGL could be affected by the Kd selected.

Power plant sites should review their soil characterization results to determine if either of these two criteria stated in DWUP-ISG-002 apply and the need for further justification of the Kd parameter evaluated.

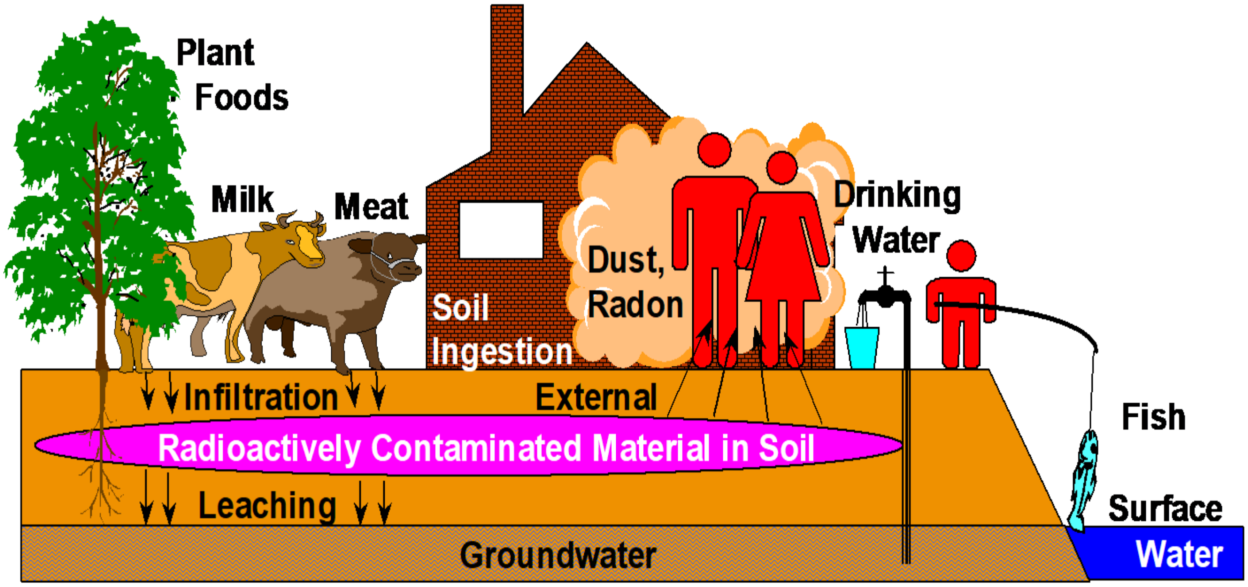

Figure 6-1: Dose Pathways of RESRAD Dose Modeling Code

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 formally part of the plant and has been released for use as an office. As with the dose from residual radioactivity in soil, the DandD and RESRAD-Build computer codes are commonly used to calculate these DGCLs. Building Occupancy DCGLs calculated using the DandD and RESRAD-Build (as used by Connecticut Yankee) code are compared in Table F-9.

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 6-3) 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. This option would also identify less likely, alternate scenarios to the reasonably foreseeable scenarios, to understand the robustness of the analysis.

In NRC Regulatory Issue Summary 2004-08, *Results of the License Termination Rule Analysis*, (Reference 6-4) the NRC Commissioners approved the recommendations of SECY 03-0069. (Reference 6-3). This allowed licensees to justify scenarios based on reasonably foreseeable future land use as opposed to defaulting to the very conservative scenarios.

The NRC based this revision on an NRC review of dose modeling experiences of the plants that had begun decommissioning prior to 2003 including Trojan, Connecticut Yankee, and Maine Yankee. Discussions on how these plants chose their land use scenarios are provided later in this chapter. Also discusses 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 NRC Policy Issue, 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 NUREG 1757, “Consolidated Decommissioning Guidance,” NUREG 1757 provides detailed guidance on the conduct of decommissioning. In 2022, the NRC issued revision 2 to NUREG 1757 (Reference 1-4). The following is a summary of the guidance in NUREG 1757, Rev 2, concerning establishment of exposure scenarios for use in dose modeling.

Table 6-1 below from NUREG 1757, Revision 2, defines the different types of scenarios that are evaluated in determining site release limits (i.e., DCGLs). These definitions are important when evaluating and using the results of dose modeling for different types of scenarios. NUREG 1757 states if peak doses from the less likely but plausible land use exposure scenarios are significantly above the dose standard, the licensee would need to provide greater assurance that the exposure scenario is less likely to occur, especially during the period of unacceptably high dose.

Table 6-1: Description and Comparison of Dose Modeling Scenario Types (NUREG 1757 Rev 2)

	Exposure Scenario Type	Description
Plausible Exposure Scenarios	*Compliance Exposure Scenarios (Results Compared to Dose Standards)*[1]	
	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	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 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).

[1] 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 for Land Areas

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 Reference 6-11, Argonne National Laboratory Presentation) depicts most of the typical exposure pathways for the Industrial Use 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

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 of 1968 and was permanently shut down in December of 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.

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 6-5).

- The critical group for site-specific analysis of the Industrial Area at Big Rock Point was a modified resident farmer who moves 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% 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.

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 Use Scenario to be used in determining the release limits for the Rancho Seco site.

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. Also discussed are dose modeling experiences at nuclear power plant sites since the implementation of the License Termination Rule in 1996.

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. Some of the options are discussed in the following sections. Additional experiences are discussed in Appendix F.

6.2.1 Options for Development of Land Area Site Release Limits

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 (Reference 1-4), Table H.2. The soil screening values for most of the radionuclides that have been found to be detected during past decommissioning projects are listed in this NUREG. These screening values were determined using the NRC DandD dose modeling code, which was developed by the Sandia National Laboratories (information on the DandD code is contained in Reference 6-1). The default parameters of the DandD 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 as long as the assumptions inherent in the screening code (given below) are met. 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 (Reference 1-4), 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 source is located at the surface and the unsaturated zone below the source and saturated zone 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.

It should be noted that revised NRC guidance in NUREG-1757, Volume 2, Revision 2, Section I.2.3.1, states that the DandD code used to develop screening values only considers surface residual radioactivity and uses a simple method to account for area on dose. The treatment of the area on dose is very different between RESRAD and DandD. Additional justification would be needed to use adjusted screening values when subsurface residual radioactivity is present because DandD is only appropriate for surface residual radioactivity and has a simplistic treatment of area on dose that differs from RESRAD.

- 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 1-4) that the use of the single default parameter set for all radionuclides could result in overly conservative limits. 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. The user is instructed in the NUREG that tailoring the default parameter set to individual radionuclides would, in most cases, result in higher DCGLs.

Adjusting NRC Screening Values for Potentially Contaminated Groundwater

Although the model used to calculate the screening DCGLs for soil discussed above include the dose from the water pathway after the beginning of the modeling period, these models to not include the dose from radionuclide contamination that is already present in groundwater. Appendix F discusses methods that can be used to calculate the dose from groundwater contamination. In a graded approach based on the extent of the radionuclide contamination, these methods range from conservatively applying the maximum level of groundwater contamination across the site to cases where it may be necessary to apply different dose values for individual survey areas/units. In this latter approach, additional data and detailed justification will be required to support the use of these different dose values.

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

The first step in adjusting the NRC Screening Values is determining the dose resulting from the contamination in groundwater. One approach to achieve this is to determine and apply pathway dose conversion factor (PDCF) for groundwater dependent pathways as is discussed in Appendix F. The experience at decommissioning plants has typically been to establish the U.S. Environmental Protection Agency (EPA) Maximum Contamination Levels (MCLs) (Reference 6-7) as limits for radionuclide contamination in groundwater. For all the sites that have achieved release from their NRC licenses to date, the EPA MCL concentrations have not been exceeded either for all monitoring wells at the site or for the average of the groundwater concentrations within the capture zone of a drinking water well of a theoretical future resident of the site.

In a related topic, Section 2.4 describes the importance of a Conceptual Site Model (CSM) for both the characterization of groundwater contamination for a site and the abstraction and development of hydrological inputs for the dose models to be used to calculate the dose from groundwater contamination. It should be noted that monitoring well concentrations may not reflect higher concentrations nearer to known or unknown sources. For sites where groundwater concentrations approach the site release limits, more sophisticated methods such as Fate and Transport Modeling may be needed to adequately assess the potential dose from groundwater at a site.

6.2.2 Options for Development of Site Release Limits for Structures

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. Some of these options evaluate building surfaces that will remain onsite and could be occupied, versus building surfaces that will be backfilled (buried) and could eventually contribute to soil and groundwater contamination. The following subsections describe a few examples of each of these cases while Appendix F describes provides more examples options including the use of NRC Screening Values for building surfaces that do not require dose modeling. It should also be noted that NRC guidance in DUWP-ISG-02 (Reference 5-6) discusses exposure scenarios and survey of substructures that are planned to be back-filled.

Building Occupancy Scenario

For most sites in the US, most licensees have used the Building Occupancy Scenario as a basis for determining DCGLs for buildings that have the potential to be occupied after the release of the site. This scenario and how DCGLs have been determined using it are discussed in this chapter and Appendix F.

As there are other configurations of buildings that could be applicable after site release, options for determining site release limits other than the Building Occupancy Scenario are needed. Some of the approaches that have been used at power plant sites are discussed in the following sections and in Appendix F.

6.2.3 Options for Development of Site Release Limits for Embedded and Buried Piping

Embedded piping is piping embedded in a durable material, typically concrete, that cannot be easily removed without significant effort and tools. The HSA and characterization surveys indicate whether residual radioactivity is present in each system and/or section of piping. The normal room surveys will adequately account for direct (external gamma) radiation from the pipes when the pipes are in place and undisturbed. The direct (external gamma) dose from the pipes will be in addition to the dose from the residual radioactivity on surfaces in the room. It may also be necessary consider building renovation that would disturb the piping, as described in NUREG/CR-5512, Volume 1, “Residual Radioactive Contamination from Decommissioning” (Reference 6-8). If this is done, the survey should be consistent with the dose modeling assumptions.

Buried piping is that which is present in the subsurface outside of a structure. As with embedded piping, the HSA and characterization surveys indicate whether residual radioactivity is present in each system and/or section of piping.

As there is no prescriptive NRC guidance for modeling the dose from embedded pipe penetrations and buried piping, licensees have proposed approaches in Chapter 6 of their LTPs. The following three sections show examples of how the Zion site addressed determining the dose from these types of commodities. Additional examples are included in Appendix F.

Zion FSS experience with embedded piping

For the Zion site, a contribution to the total Subsurface Structure Dose Model (SSDM called BFM in their LTP) dose needed to be calculated for embedded piping. The SSDM 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 (these were called $DCGL_{EP}$). $DCGL_{EP}$ were calculated for each of the embedded pipe survey units. The Base Case $DCGL_{EP}$ values from Zion LTP are reproduced in 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 6-2: Base Case DCGLs for Embedded Pipe (DCGLEP)

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

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 was included in the survey units for both basements. The residual radioactivity in the penetration is assumed to be released to both basements simultaneously.

The SSDM 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 (these are called $DCGL_{PN}$). 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 $DCGL_{PN}$ 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 $DCGL_{PN}$ values from the Zion LTP (Reference 5-2) are reproduced in Table 6-3 (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 $DCGL_{PN}$ values in Table 6-3to account for the dose from the eliminated IC radionuclides.

Table 6-3: Base Case DCGLs for Penetrations (DCGLPN)

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

Zion FSS experience with buried pipe

For the Zion site, buried pipe is defined as pipe that runs through soil. The critical group for the buried piping dose assessment is the Resident Farmer.

The buried pipe DCGLs (these are called $DCGL_{BP}$), in units of dpm/100cm^2, were determined for two scenarios:

- assuming that all pipe is excavated and
- assuming that all pipe remains in situ.

Although unrealistic, for the purpose of the bounding modeling approach used, the dose from the two scenarios is summed to determine the $DCGL_{BP}$. RESRAD was used to calculate $DCGL_{BP}$ for both the excavation and in situ buried pipe scenarios using the parameters developed for soil modified as necessary for the buried pipe source term geometry. Details on dose assessment methods are provided in the Zion LTP (Reference 5-2). A brief overview of scenario assumptions is provided below.

The excavation scenario assumes that all buried pipe is excavated after license termination and all activity on the internal surfaces of the pipes is instantly released and mixed with surface soil. The in situ scenario assumes that all of the buried piping remains in the “as-left” condition at the time of license termination and that all activity is instantly released to adjacent soil. Two separate in situ calculations were performed. The first calculation assumes that all pipes are located at 1 m below the ground surface in the unsaturated zone and the second assumes that all pipes are located in the saturated zone. The lowest in situ DCGL from either the 1 m deep unsaturated or saturated scenario was assigned as the in situ $DCGL_{BP}$.

6.2.4 Options for Development of Site Release Limits for Buried Materials

Appendix J of NUREG 1757 Vol 2 Rev 2 (Reference 1-4) provides guidance on the important considerations and conceptual scenarios that should be considered when developing site release limits for buried materials. This section will summarize the information in this Appendix of NUREG 1757. Examples of how licensees have determined site release limits that address these considerations are included in Appendix F.

To develop the exposure scenario(s) for the critical group, the analyst should address the following questions:

- How does the residual radioactivity move through the environment?
- Where can humans be exposed to the environmental radiological concentrations?
- What are the exposure group's habits that will determine exposure?
  - What do they eat and where does it come from?
  - How much do they eat? Where do they get water and how much water do they drink?
  - How much time do they spend on various activities?

In the absence of site-specific information, residential scenarios are generally assumed in the following examples, which are provided in NUREG 1757 for illustrative purposes, although arguments can be presented for why other scenarios are reasonably foreseeable and residential scenarios are less likely but plausible (or implausible). For more information on developing exposure scenarios as summarized in the next two sections, see Appendix I and Appendix M of NUREG 1757 Volume 2 (Referene 1-4).

Buried Material Conceptual Models, Exposure Scenario, Exposure Pathways, and Critical Group

For sites with buried material, one conservative conceptual model could assume that all the buried radioactive material at depth was instead at the surface. That is, this scenario takes no credit for the clean cover (the clean cover is removed in the conceptual model) and models assume:

- leaching of the radionuclides into the groundwater which is then used by a resident, as well as
- considering direct radiation from exposure to the soil at the surface.

Care should be taken to ensure the modeled vadose (unsaturated) zone thickness is equal to or less than the actual vadose zone thickness (i.e., the contamination zone should not be moved to the top of the cover but should remain at the actual elevation).

Alternatively, credit for the cover may be taken and two exposure scenarios can be developed:

- leaching of the radionuclides from their buried position to groundwater, and
- intrusion into the buried residual radioactivity by basement construction (or other large-scale excavation scenario) or well drilling.

In the second scenario the displaced soil, which includes part of the residual radioactivity, is spread across the surface. All the exposure pathways are considered for the second scenario involving intrusion into the buried residual radioactivity and, although not all the source term is in the original position, leaching will occur both from the remaining buried residual radioactivity (if there is any) and the surface soil.

Additional Considerations for Backfilled Basement Conceptual Models, Exposure Scenarios and Pathways

For backfilled basements, a similar approach can be considered as that for buried material. An intrusion event (well-driller scenario) that includes leaching into the groundwater can be developed. In the well-driller scenario, the backfilled basement is assumed to remain undisturbed except for a well drilled through a portion of the backfilled substructure for drinking water. The well-drilling scenario considers:

- pathways associated with use of contaminated groundwater from an onsite well, and
- direct (and indirect) exposure to drilling spoils that are brought to the surface during the installation of the onsite well.

These two exposure scenarios have different conceptual models. For the groundwater scenario, licensees may be able to make simplifying assumptions if the simplifications do not underestimate the potential dose (e.g., licensees may be able to assume all the residual radioactivity exists as a contaminated zone layer of soil at the elevation of the floor of the underground structure closer to the water table aquifer). To simplify the conceptual model further, the concrete floor and walls can be assumed to not affect flow, allowing more rapid transport of residual radioactivity to groundwater. If the simplifications lead to excessive pessimism, licensees have the option to model the actual configuration of residual radioactivity, degradation of cementitious materials, and flow and transport of residual radioactivity to groundwater. However, given the complexity and uncertainty in more realistically modeling these processes, it is expected that uncertainty will likely need to be managed with conservative assumptions. The licensee should communicate with the NRC early in the process to ensure that a technically acceptable approach is developed for these more complex problems.

If the concrete floors and walls are assumed to be intact, then water could fill the basement and cause contamination of near-surface soils. Depending on the depth of the buried residual radioactivity below ground surface and mechanisms for upward transport, residual radioactivity transported near the surface could present additional exposure pathways and scenarios that need to be considered (e.g., exposure to residual radioactivity transported to the surface and/or exposure to residual radioactivity from basement excavation scenario if radioactivity below 3 m (10 ft) of the surface can be transported within 3 m (10 ft) of the surface). Licensees could take credit for measures to ensure that water does not fill up in the basement to eliminate this scenario from consideration. Additionally, licensees can take credit for cover materials and depth of the residual radioactivity in eliminating surface exposure pathways. The adequate depth will depend on site–specific information (e.g., water table level and variations in the water table) and will need to be evaluated on a case-by-case basis.

For the well-driller direct exposure pathway, a well is assumed to be installed through the fill material, and drilling spoils along with a portion of the concrete floor are brought to the surface. The well installation may occur before any leaching from concrete occurs. The residual radioactivity in the concrete of the basement floor that is contacted by the borehole during installation of the well is assumed to be mixed with the column of drilling spoil fill material above the floor surface, brought to the ground surface, and spread.

Additional scenarios that may need to be considered for large backfilled subgrade structures (e.g., containment basements, auxiliary building basements, and/or turbine basements at a reactor site) may be described as large-scale excavation. A large-scale excavation scenario, which would bring up more residual radioactivity compared to a well-driller scenario, may need to be considered given the depth, geometry, and materials associated with the source. In this scenario, a portion or all the remaining backfilled basement walls and/or floor are removed in the future after license termination. In this scenario, all the residual radioactivity is assumed to remain on the concrete to maximize the concentrations in the concrete brought to the surface. The dose to a construction worker, disposal facility worker, or other member of the public who may be exposed to the residual radioactivity on the excavated concrete and/or fill material would need to be considered. The radionuclide concentrations (Bq/kg or pCi/g) in the excavated concrete are directly related to the ratio of concrete surface area to concrete volume excavated. For substructures that have thick walls and for where the residual radioactivity is limited to the surface, assuming a partial excavation that includes only the walls with the minimum thickness would result in a higher concentration. Arguments could be presented by the licensee why a large excavation scenario is not reasonably foreseeable or why certain pathways are less likely but plausible, or implausible and are simply used to inform decision-making, or do not need to be considered. See Table 6-1 for additional information on compliance, informative, or eliminated exposure scenarios).

It should be noted that a simplistic conservative approach for the well drilling and excavation scenarios may be convenient to apply at some sites. This approach would assume that assume the residual radioactivity is located at the surface (no clean cover) and calculate DCGLs (or assess dose) from both surface and in situ groundwater leaching scenarios simultaneously as described in NUREG-1757, Volume 2, Revision 2, Appendix J.

Use of RESRAD for Modeling of Buried Materials

Appendix J of NUREG 1757 provides guidance on using the RESRAD family of dose modeling codes to determine pathway dose conversion factors for the scenarios described in this subsection. Interim Staff Guidance (ISG) in *Radiological Survey and Dose Modeling of the Subsurface to Support License Termination* (Reference 5-6) supplements NUREG-1757, Volume 2, and provides guidance on radiological survey approaches for substructures as well as limitations of codes such as RESRAD-ONSITE (see Section 3 of the ISG) in assessing groundwater dependent pathway doses for submerged sources such as reactor basement substructures. A summary of the key points of the ISG concerning these limitations is included in Appendix F.

6.2.5 Recent Example of Dose Modeling Scenarios

Dose Modeling Overview for Fort Calhoun Station

As of the issuance of this revision of this document in 2024, the Fort Calhoun Station site provides a recent example of the dose modeling scenarios that have be investigated at a site. The following is a summary of the information on this subject in the Fort Calhoun LTP. Table 6-4 provides the foreseeable land use scenarios and exposure pathways investigated while Table 6-5 provides the exposure pathways that are applicable for the compliance scenarios (including those for structures and buried materials).

Table 6-4: Reasonably Foreseeable Land Use Scenarios, Critical Groups. and Pathways

Environmental Pathway	Exposure Pathway	Land Use Scenario					
		Residential Farming	Industrial (Light Industry)	Industrial (Commercial Agriculture)	Residential	Urban Residential	Recreational
Direct Exposure	External Radiation	ü	ü	ü	ü	ü	ü
Airborne (Particulate, H-3)	Inhalation	ü	ü	ü	ü	ü	ü
Plant Foods	Ingestion	ü			ü		
Livestock - Meat	Ingestion	ü					
Livestock - Milk	Ingestion	ü					
Onsite Groundwater	Ingestion	ü	ü	ü	ü		ü
Direct Soil Contact	Ingestion	ü	ü	ü	ü	ü	ü
Critical Group		Adult that resides onsite and derives a large fraction of annual food intake from onsite agriculture, livestock.	Adult that works onsite full time, pre-dominantly indoors, performing light industrial activities.	Adult worker that periodically occupies the onsite area which is part of a larger commercial farming operation	Adult resident that derives a small fraction of annual food intake from an onsite garden. No livestock raised onsite.	Adult resident that does not maintain a garden or livestock. Drinking water is supplied by an offsite source	Adult that periodically accesses the site (parkland or recreation area) for hiking, camping, etc. No food is harvested from the site; onsite well is drinking water source.

Table 6-5: Compliance Scenario Environmental Pathways and Exposure Pathways

No.	Source	Environmental Pathway(s)	Exposure Pathway
1	Soil	resuspension of soil into air (airborne particulate)	Inhalation
2	Soil	onsite direct exposure	External radiation
3	Soil	soil to edible plant	Ingestion of plants

No.	Source	Environmental Pathway(s)	Exposure Pathway
4	Soil	soil to forage plant à beef cow	Ingestion of meat
5	Soil	soil to forage plant à dairy cow	Ingestion of milk
6	Soil	soil to groundwater à well à plant by irrigation	zIngestion of plants
7	Soil	soil to groundwater à well à forage by irrigation à beef cow	Ingestion of meat
8	Soil	soil to groundwater à well à forage by irrigation à dairy cow	Ingestion of milk
9	Soil	soil to groundwater à well à drinking water	Ingestion of drinking water
10	Soil	soil to hand à mouth	Ingestion of soil
11	Buried Pipe	activity from internal surface of buried pipe to subsurface soil à environmental pathways 2-9	Direct exposure, ingestion of plants, meat, milk, and drinking water
12	Buried Pipe	excavation of buried pipe à activity from internal surface of buried pipe to surface soil à environmental Pathways 1-10	Exposure Pathways 1-10
13	Backfilled Basement Surfaces (walls/floors)	release from concrete to fill material à equilibrium desorption from fill to pore water à pore water to well à environmental pathways 2-9 (with fill replacing soil)	Direct Exposure, ingestion of plants, meat, milk, and drinking water
14	Backfilled Basement Surfaces (walls/floors)	basement concrete incorporated with drilling spoils during well installation à drilling spoils to ground surface and treated as soil à environmental pathways 1-10	Exposure Pathways 1-10
15	Backfilled Basement Concrete	excavation of basement concrete à concrete debris spread on ground surface à concrete debris treated as soil à environmental pathways 1-10	Exposure Pathways 1-10
16	Basement Embedded Pipe	activity from internal surface of embedded pipe released to fill à equilibrium desorption from fill to pore water à pore water to well à environmental pathways 6-9 (with fill replacing soil)	Ingestion of plants, meat, milk, and drinking water
17	Existing Groundwater	groundwater to well à well water pathways 6-9	Ingestion of plants, beef, milk, and drinking water
18	Basement Fill	equilibrium desorption from fill to pore water à pore water to well à environmental pathways 2-9 (with fill replacing soil)	Direct Exposure, ingestion of plants, meat, milk, and drinking water

6.2.6 Less likely But Plausible Exposure Scenarios for Fort Calhoun

There are two exposure scenarios that are categorized as less likely but plausible (LLBP) for Fort Calhoun. The first is exposure to contaminated drilling spoils brought to the ground surface during installation of a water well through the Auxiliary Building or Turbine Building basement that contacts an embedded pipe. The receptor is the resident farmer AMCG. The dose to a worker will be less due to decreased occupancy time and elimination of food pathways. Note that exposure to drilling spoils where the source term is contaminated concrete is included as a compliance scenario at Zion.

The second LLBP exposure scenario is offsite processing/recycling/disposal of excavated basement concrete and Containment Building steel liner. The offsite processing/recycle/disposal dose is calculated using the dose factors for concrete and steel in NUREG-1640, “Radiological Assessments for Clearance of Materials from Nuclear Facilities,” Table 2.1. There are multiple exposure scenarios (and receptors) evaluated in this such as dismantling and handling while decommissioning, exposure to waste piles, transport to the recycle or disposal site, disposal at landfill, and commercial reuse of the materials. The results of this second LLBP scenario was that the dose would be less than the dose calculated for the subsurface structure dose model (SSDM Called BFM in their LTP) wall/floor compliance scenario which is discussed in Appendix F.

6.3 References

6-1 Oak Ridge Associated Universities, https://ramp.nrc-gateway.gov/codes/resrad/download

6-2 Argonne National Laboratory, https://resrad.evs.anl.gov/download/

6-3 NRC Policy Issue, SECY-03-0069, Results of the License Termination Rule Analysis, May 2, 2003.

6-4 NRC Regulatory Issue Summary 2004-08, “Results of the License Termination Rule Analysis,” May 28, 2004

6-5 Big Rock Point Restoration Project License Termination Plan, Revision 0, April 1, 2003.

6-6 LaCrosse Boiling Water Reactor License Termination Plan, Chapter 6, Revision 1, May 2018

6-7 EPA Document # 816-D-00-002, “Implementation Guidance for Radionuclides,” January 2002, Appendix I has the MCLs.

6-8 NUREG/CR-5512, Volume 1, “Residual Radioactive Contamination from Decommissioning,” October 1992

6-9 NUREG-1640, “Radiological Assessments for Clearance of Materials from Nuclear Facilities,” June 2003.

6-10 Fort Calhoun Station License Termination Plan, Revision 1, December 2023.

6-11 Argonne National Laboratory Presentation at EMRAS II NORM and Legacy Sites WG Meeting, “RESRAD Offsite Code and Derivation of Cleanup Criteria,” January 27, 2010.

6-12 NRC Letter to B. Mongomery (NEI), U.S. Nuclear Regulatory Commission Observations and Insights of Nuclear Energy Institute (NEI) Technical Report NEI 22-01, "License Termination Process" dated April 30, 2024. (ML24039A183)

6-13 NUREG/CR 7267 "Default Parameter Values and Distributions in RESRAD-ONSITE V7.2, RESRAD-BUILD V3.5, and RESRAD-OFFSITE V4.0 Computer Codes," May 2024.

7 UPDATE ON SITE-SPECIFIC DECOMMISSIONING COSTS

7.1 Introduction

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 site-specific DCE cost estimate should be adjusted for present value if inflation and disposal costs have changed. However, the licensee must continue to provide the annual decommissioning funding assurance report under 10 CFR 50.75, “Reporting and Recordkeeping for Decommissioning Planning,” until the license is terminated.

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.

Any significant changes to the remaining decommissioning or site restoration costs or significant changes to the schedule should be reflected in an update to this chapter of the LTP and in the PSDAR. Significant is defined as any change in cost or schedule that is more than 15% of the total decommissioning cost or greater than 15% of the entire decommissioning schedule for the nuclear plant.

An early submittal of the LTP may require an update to this chapter if significant delays are encountered during the plant decommissioning process. An update will also likely be required when the DOE schedule to remove the spent fuel and GTCC waste from the site is known.

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. RG 1.159, “Assuring the Availability of Funds for Decommissioning Nuclear Reactors,” provides detailed guidance on methods for estimating decommissioning costs and on financial assurance mechanisms. 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. Regulatory Guide 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," 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.

7.2 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 present a description of how the cost estimate was prepared and a summary and breakdown of the estimated costs.

7.2.1 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.

7.2.2 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.

7.2.3 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.

7.2.4 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 are typically presented in the LTP.

7.2.5 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.

7.2.6 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, DOE G 430.1-1, 3-28-97 (DOE G). 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."

7.3 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).

7.4 References

7-1 Regulatory Guide 1.159, Revision 1, “Assuring the Availability of Funds for Decommissioning Nuclear Reactors,” October 2003.

7-2 NUREG-1307, “Report on Waste Burial Charges: Changes in Decommissioning Waste Disposal Costs at Low-Level Waste Burial Facilities, Revision 20,” January 2025.

7-3 DOE Cost Estimating Guide, “DOE G 430.1-1,” March 28, 1997.

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. This supplement provides the 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.

The Generic Environmental Impact Statement (GEIS) reference facilities were developed to broadly and generically represent categories of licensee facilities. Specific facilities will not exactly match the descriptions of the reference facilities. The primary purpose of comparing a specific facility to the reference facility with regard to dose assessment is to determine whether the specific facility has important contaminants, potential scenarios, or pathways that were not analyzed for the reference facilities or which may be sufficiently different from those in the GEIS to change conclusions regarding environmental impacts. In general, if a specific facility has contaminants, concentrations, and special distributions less than or generally equivalent to those used for the reference facilities, the GEIS should be applicable. Additional guidance is found in Appendix A to NUREG-1748, “Environmental Review Guidance for Licensing Actions Associated with NMSS Programs,” August 2003.

The focus of the supplement to the environmental report is the environmental impacts of decommissioning and reasonable alternatives. The Safety Evaluation Report (SER) evaluates and documents the safety aspects and compliance with NRC regulations. The safety and environmental reviews are conducted in parallel. Although there is some overlap between the content of an SER and the environmental report overlap, the intent of the documents is different. Much of the information describing the affected environment is also applicable to the SER (e.g., traffic patterns, demographics, geology, and meteorology) and the licensee should ensure consistency between the supplement to the environmental report and the SER.

Licensees need to follow the guidance found in Regulatory Guide 1.179, “Standard Format and Content of License Termination Plans for Nuclear Power Reactors,” Revision 2, July 2019 and NUREG-1748, “Environmental Review Guidance for Licensing Actions Associated with NMSS Programs,” August 2003 when drafting the supplement to the Environmental Report. These documents provide the latest NRC guidance.

The NRC staff are required to consult with other federal, state, local and tribal agencies during their review of this chapter of the LTP. These consultations should be initiated early in the review process as it may take quite some time for the other agencies consulted to respond to the NRC. Also, any additional requirements imposed by other agencies in the course of the environmental review should be known early on in order to prevent delays in completing decommissioning due to additional required environmental activities. The NRC staff will likely consult with other agencies on the Endangered Species Act of 1973 (ESA, 16 U.S.C. § 1531 et seq.), the Magnuson-Stevens Fishery Conservation and Management Act (MSA, 16 U.S.C. § 1801 et seq.), as amended by the Sustainable Fisheries Act of 1996 (Public Law 104-267), the National Marine Sanctuaries Act (NMSA, 16 U.S.C. § 1431 et seq.), and Section 106 of the National Historic Preservation Act of 1966. The NRC guidance on how these consultations are performed is provided in NUREG-1555, “Standard Review Plans for Environmental Reviews for Nuclear Power Plants, Supplement 1: Operating License Renewal, Appendix A, Interagency Consultations for Ecological Resources, and Appendix B, National Historic Preservation Act Section 106 Review and Consultation,” Revision 2. As stated in Chapter 1, a pre-application meeting with the NRC environmental staff is recommended for this section of the LTP.

8.3 Lessons Learned

While licensees have generally met these requirements, there are a few lessons learned that should be considered when drafting this chapter of the LTP. Each environmental issue is assigned to be generic or site specific. The site-specific issues require an additional site-specific review. When performing the site-specific review, current environmental information must be used, and a licensee cannot rely on past environmental assessments to conclude there is no change to the environmental impact. The following discussions reflect lessons learned from current or past decommissioning projects.

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

It should be noted that Appendix A, “Interagency Consultations for Ecological Resources,” in NUREG-1555, “Standard Review Plans for Environmental Reviews for Nuclear Power Plants, Supplement 1: Operating License Renewal,” Revision 2 contains information on:

- The ecological consultations that NRC may be required to conduct or what statutes require such consultations
- The types of information environmental reports should contain to facilitate and support each consultation
- the permits and authorizations that may result from these consultations

Although this document is tailored to operating reactor license renewal, the guidance pertaining to ecological consultations is relevant to any NRC action.

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 are 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 nesting and offspring. Close communication with state and federal environmental agencies is critical so as not to delay the project. Further guidance on how the NRC consults with other Federal and State agencies on threatened and endangered species is provided in NUREG-1555, “Standard Review Plans for Environmental Reviews for Nuclear Power Plants, Supplement 1: Operating License Renewal, Appendix A, Interagency Consultations for Ecological Resources, ,” Revision 2.

8.8 Environmental Justice

Environmental Justice information is no longer required in the environmental review for decommissioning. Therefore, this information does not need to be included in the LTP application.

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 communications with the SHPO are important to ensure all stakeholders agree as to the actions taken. Further guidance on how the NRC consults with other Federal and State agencies on cultural and historic activities beyond the operational area is provided in NUREG-1555, “Standard Review Plans for Environmental Reviews for Nuclear Power Plants, Supplement 1: Operating License Renewal, Appendix B, National Historic Preservation Act Section 106 Review and Consultation,” Revision 2.

8.10 References

8-1 NUREG-1555, "Standard Review Plans for Environmental Reviews for Nuclear Power Plants, Supplement 1: Operating License Renewal, Appendix A, Interagency Consultations for Ecological Resources, and Appendix B, National Historic Preservation Act Section 106 Review and Consultation," Revision 2.

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 (Reference 1-3). This NUREG references another NRC guidance document (NUREG-1757, Vol. 2, Reference 1-4) 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, Reference 1-1). 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."

Despite guidance provided in various NRC documents, there has been some level of confusion as to what constitutes a "stand-alone document." For example, if all instrumentation and laboratory raw data (e.g., gamma-spectroscopy reports) are included for each survey unit, the FSS reports can often contain thousands of pages. 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
- This NEI guidance document (NEI 22-01) provides recommendations for information to be included in FSS reports in order to consolidate the amount of information included which will serve to 1) provide the licensee with the information required to develop a concise and cohesive report and 2) Reduce the amount of time required for NRC review.

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.

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 $DCGL_W$

  - a statement that a given survey unit satisfied the $DCGL_W$ and the elevated measurement comparison if any sample points exceeded the $DCGL_W$

- 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 4-1.

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

Other information may include:

- What existed in the survey unit

- What radiological operations occurred in the survey unit

- What remediation was performed or what structures removed

- Were any DRPs identified during FSS

- Was a FSS failed and had to be reperformed, and if so, what was the scope of any post FSS remediation

- Did scanning identify significant elevations that required investigation

- Were any elevations identified

- Are there any unusual characteristics that the reviewer should be aware of (e.g., two surveys reported for a survey unit...one of bottom of excavation and another of top of backfill), etc.

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 of this volume for information on survey data quality and reporting,
- Section A.9 from Appendix A of this volume 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 is 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 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
- Questions 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 Shutdown 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 DandD 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 have the opportunity to raise issues/concerns prior to or during the implementation phase of FSS.

It should be noted that NRC decommissioning inspection procedures are revised periodically. Licensees should be vigilant of changes to the NRC procedures that involve the license termination process and revise their process/procedures as needed to reflect changes to NRC inspection procedures.

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 some areas for gains in efficiency in the confirmatory survey process.

- While a confirmatory survey attempts to replicate the licensee's FSS design as described in the approved licensee LTP(i.e., scan speed, sample density, etc.) it is not necessary to follow all aspects of the survey design. For example, if discrete objects or very small, elevated areas are found, confirmatory survey personnel may elect to slow the scanning speed and adjust detector height to ensure better sensitivity. They may also elect to use traditional handheld detectors in lieu of in-situ gamma spectroscopy to characterize a structure.

- 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, 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 necessarily limited to inspections, confirmatory surveys, and SERs for any safety evaluation reports that may have addressed the FSS plan.” To further strengthen the technical review process, the licensee should request that technical reviewer review the above cited documents, and also have a familiarity with the plant through either site visits or a presentation on the site decommissioning history. Knowing the site history, the oversight of decommissioning, and the confirmatory measurements should allow the review of submitted data and associated analysis with a minimum of raw data and process description as required by the LTP and procedures.

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. MARSSIM Revision 2 defines conditions that must be met for advanced technologies to be employed as the only survey technique. These are defined in Section 5.2.1.3 and include:

- 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 an advanced survey technology or where the MDC is too high 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 as long as the MDC is less than the site release limits. The following provides some examples of advanced technologies.

It should be noted that, if novel technology is planned for use for surveying or otherwise in the decommissioning process, the licensee should consider direct comparisons to traditional methodologies (e.g., sampling, handheld detector use, etc.). A white paper or topical report on the technology or methodology should be submitted to the NRC as early in the process as possible because NRC understanding and acceptance may require additional research and verification.

A.1 Position Sensitive Proportional Counters

The Surface Contamination Monitor (SCM), originally developed by Shonka Research Associates, Inc. creates a spatially-correlated log of the measurements made as the detector is passed over an area. Shonka developed the automated SCM along with a generalized data management system with U.S. NRC funding. The SCM uses a patented position-sensitive proportional counter (PSPC) that can be fabricated in various lengths (typically 0.25 to 2.5 meters). Multiple detectors can be connected together in a variety of ways to:

- create a larger array or to simultaneously use a second detector with a beta filter to measure gamma background when background is spatially variable, or
- use coincidence counting logic for low level alpha detection.

Extensive improvements have been made to the detectors, the electronics systems, the operating software and the post processing software over the decades of use. When tested by DOE in the Large Scale Demonstration Projects (LSDP) program, the dominant feature noted was the speed of surveying and reporting the data. This speed was important during the major DandD efforts that utilized SCM and SIMS, which often were ones where the flat areas to be surveyed were large.

The logging of all measurements by the SCM allows quantitative assessments of activity levels to be made, thus serving the same role as fixed measurements. Having all of the measurements logged allows statistical analyses to be made using a large number of samples, which provides for enhanced detection sensitivity relative to traditional scanning. The sensitivity achieved using these advanced survey methods may, in some cases, be small enough relative to the site release limits that the advanced method alone will allow a decision to be made as to whether a survey unit meets the release criterion without the need for additional fixed measurements (thereby saving the cost of performing fixed measurements). The fact that the instrument records every measurement made over the entire area it covers with no blind spots inherently addresses the issue of small areas of elevated activity. Average and maximum residual activity concentrations can be quantified over any area desired, allowing one to assess compliance with the site release limits by inspection.

A.2 In-Situ Gamma Spectroscopy

In-situ gamma spectrometry is an established technique for assaying the average radionuclide concentration in large volumes of material such as activated concrete. It has the advantage of being able to assess large areas (typically 28 m^2) with a single measurement. If desired, the detector's field of view can be reduced through collimation or reduced distance to the target area to allow assay of smaller areas and/or to reduce the MDC achieved.

In situ object counting in this context refers to gamma spectrometry systems that use high-purity germanium (HPGe) or scintillation (LaBr, NaI, CeBr) detectors and include software capable of modeling photon transport in complex geometries for the purpose of estimating detector efficiencies. This eliminates the need for a calibration geometry representing the object to be counted. Such systems are also useful for assaying piping and complex components such as heat exchangers.

The use of in-situ gamma spectroscopy in place of scanning has been more widely used in decommissioning FSS as experience has grown. However, recognition of the following limitations should be considered. In-Situ Gamma Spectroscopy should not be considered where past remediation for DRPs or expectation for potential DRPs exist. Additionally, as indicated activity in a survey unit approaches the DCGL, complementary scanning and sampling should be considered. The following is an example of in-situ gamma spectroscopy being used in place of conventional scanning (Reference A-1).

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 consisted 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 in order 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 a number of 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 a number of smaller rooms.

A.2.2 Other Applications of In-Situ Gamma Spectroscopy

Many sites have also used in-situ gamma spectroscopy for FSS of land surveys. This method allows relatively large areas to be assessed with one count in place of scanning the area. In-situ can be used to determine if detectable soil contamination exists in a land area or to survey the inside of an excavation otherwise inaccessible from a personal safety standpoint. It has been used at some sites along with soil sampling to show compliance with the site release criteria during the FSS.

Due to the self-shielding provided by the soil being surveyed, in-situ gamma spectroscopy is generally limited to assessing contamination in the first 15 cm of soil depth within the field of view of the detector. Also, prior to its use, calculations need to be performed to ensure that the survey will be sufficiently sensitive to detect small areas of high contamination as well as distributed areas of low contamination. As in-situ gamma spectroscopy can only detect gamma radioactivity, other methods such as sampling are needed to determine if Hard to Detect Nuclides (HTDN) are present and to allow the calculation of scaling factors so that the impact of any HTDN can be factored into the assessment of an area of soil.

This section will discuss an example of the use of in-situ gamma spectroscopy for the characterization and FSS of land areas***.***

A.3 Final Status Survey of Plant Effluent Water Course: Class 2, Survey Unit #1

The area called the Plant Effluent Water Course 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

A.4 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-1. Of interest to this report are the MDCs achieved during this survey. The average MDCs were 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.

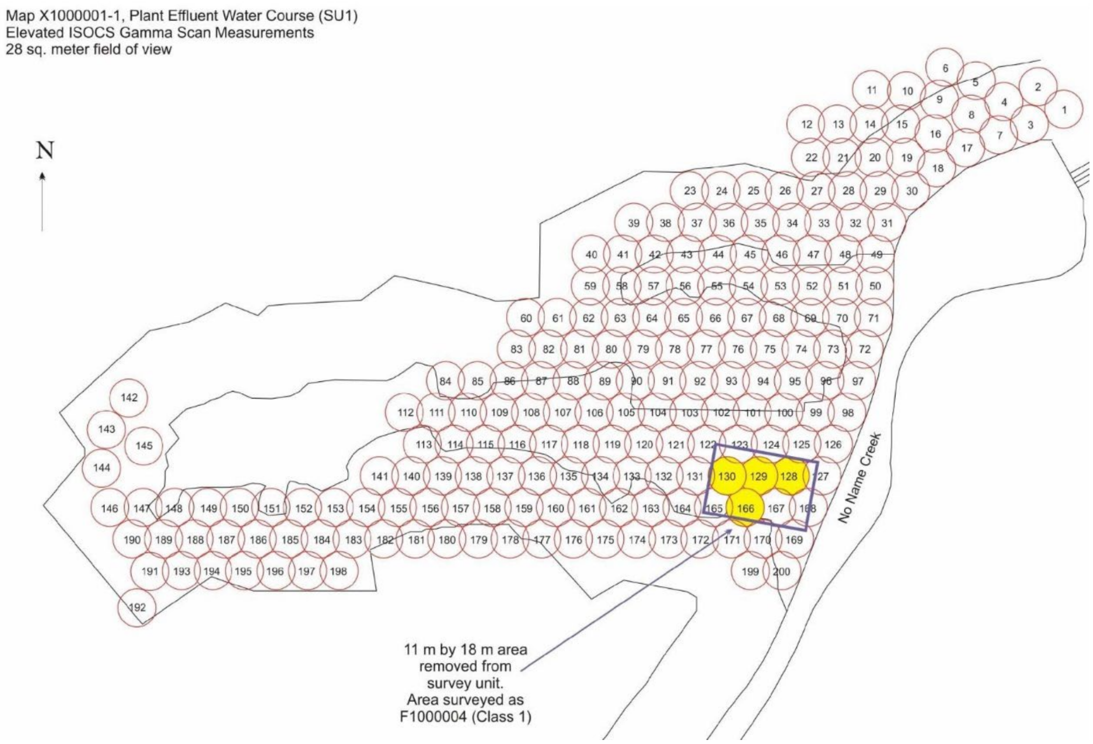

Figure A-2: Locations of In-situ Gamma Spectroscopy Counts at Rancho Seco Plant Effluent Water Course

A.5 EPRI Autonomous Survey System Demonstrations

The Electric Power Research Institute (EPRI) has performed two projects to demonstrate autonomous site characterization/final status survey systems. The aim was to demonstrate the effectiveness of autonomous survey systems seeking to reduce the time and labor required for these repetitive and labor-intensive surveys, thus leading to cost savings and improved efficiency. By gathering lessons learned, the development and deployment of future autonomous survey systems was pursued.

A.5.1 Phase 1 project: EPRI Autonomous Floors and Ground Characterization Vehicle

In October 2019, demonstration of a self-developed autonomous site characterization vehicle took place at the permanently shutdown Kewaunee nuclear power plant (Reference A-2). This vehicle could accurately follow a path while gathering and storing data, including the locations of measurements. For navigation, the vehicle used LIDAR for obstacle avoidance and GPS for outdoor localization and positioning. In indoor, GPS-denied areas, it relied on LIDAR and wheel odometry for localization. An onboard computer handled all necessary computing tasks, such as localization, navigation, measurement device communication, data collection, and storage. Additionally, it allowed remote access through a Local Area Network (LAN) and a Wi-Fi access point attached to the platform.

The radiation detection and analysis system integrated with the EPRI vehicle included equipment provided by Mirion Technologies. This system featured two detector channels, each consisting of a NaI 3x3 inch scintillation detector with LED gain stabilization and an Osprey Multi Channel Analyzer to convert detector signals to energy spectra. A Data Analyst automatically received the spectra, analyzed them, and generated radionuclide activity values every few seconds. An EcoGamma dose-rate sensor recorded the background dose rate at the vehicle locations.

During the demonstration, radioactivity spectra were collected every 3 seconds and immediately analyzed by the Data Analyst. Although the results were available in real-time and could have been transmitted via Ethernet to the vehicle computer, it was decided that real-time data usage was unnecessary for this project. To properly place the spectral assay results on the map, both the vehicle transit map data and the measurement results were saved in time-stamped files, which were then correlated to create heat maps showing radioactivity levels in the areas transited.

This demonstration showcased the feasibility to (Reference A-2):

- Identify elevated areas of activity on floors and in open land areas with low background radioactivity.
- Perform radionuclide specific analysis of detector counts allowing the determination that some of the detected activity was naturally occurring.
- Successfully map and then autonomously navigate inside a nuclear power plant.
- Autonomously travel in outdoor areas on navigations paths plotted on GPS maps downloaded from the internet.
- Achieve adequate detections sensitivities for the characterization surveys and potentially Final Status Surveys depending on the level of background present in the survey area.

A.5.2 Phase 2 project: Createc and GRI Autonomous Building Surfaces Characterization Vehicles

In late 2021 and early 2022, two demonstrations were conducted with two prototype systems developed by:

- Createc Ltd. of Cockermouth, UK, which tested their system in two configurations within their own facilities: i) vertical orientation for surveying walls from one to three meters (3.28 to 19.68 ft) above the floor, and ii) horizontal configuration for surveying the lowest one-meter (<3.28 ft) of the wall and a two-meter-wide (6.56 ft) strip of the floor.
- Gamma Reality Inc. (GRI) tested a system at their facility in Richmond, California that used an autonomous vehicle and scissor lift system to move the detector along the wall to be surveyed up to a height of three meters.

Both systems were initially driven manually to create a 3D map of the room to be surveyed. The navigation system was then programmed with waypoints for autonomous movement along the survey area. A key difference between the demonstrations was that Createc's area was additionally scanned with a fixed scanner, generating higher fidelity 3D images due to higher point cloud density. In contrast, GRI's system used a scissor lift to incrementally move the detector up for each pass, scanning the wall surface area with a few passes.

The key results of the demonstrations aimed to support suppliers to improve future site characterization or final status survey deployments through the deployment of autonomous survey systems. The two independent demonstrations showcased that:

- Both systems could autonomously survey the test area surfaces after program initiation.
- Both systems analyzed detector data to provide radionuclide-specific concentrations for Co-60 and Cs-137, avoiding manual investigations triggered by naturally occurring radionuclides.
- Both systems achieved Minimum Detectable Concentrations well below those typically required for Final Status Surveys (e.g., 50% of site release limits).
- Both systems detected sources that were a fraction of typical site release limits while scanning up to 36 times faster than conventional manual scanning and 8 times faster than in-situ gamma spectroscopy.
- Both systems located elevated areas of activity and displayed them on 3D images of the survey area. The GRI system also showed source locations on camera images of the scanned area.

A.6 References for Appendix A

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

A-2 EPRI Report # 3002018420, "Design and Demonstration of an Autonomous System for Radiological Characterization," August 2020.

A-3 EPRI Report # 3002023966, "Design and Demonstration of an Autonomous System for Radiological Characterization of Building Surfaces," October 2022.

APPENDIX B. EXAMPLE CALCULATIONS FOR BASE CASE AND OPERATIONAL DCGLS

B.1 Connecticut Yankee Experience for Land Areas

As discussed above, Connecticut Yankee (CY) expressed the total dose contribution for land areas to be from three (3) 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 was 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 (3) 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 (3) components and the total dose.

$$H_{\mathrm{Total}} = H_{\mathrm{Soil}} + H_{ExistingGW} + H_{FutureGW}$$ **Equation B-1**

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, 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 Equation B-1:

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

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 B-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 considered to be Easy to Detect (ETD). The preferred result is by alpha spectroscopy when both analyses are performed.

APPENDIX C. EXAMPLE OF TYPE OF CROSSWALK BETWEEN LICENSE TERMINATION PLAN AND REGULATORY GUIDANCE DOCUMENTS

(Note that this crosswalk is only an example of the type of crosswalk that can be used and does not include all the regulatory guidance applicable to the preparation of a License Termination Plan)

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP (page and paragraph)
1	Chapter 1, General Information	RG 1.179	The licensee’s name, address, license number, and docket number should agree with the most recent license.		
2	Chapter 1, General Information	RG 1.179	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.		
3	Chapter 1, General Information	RG 1.179	These are the following seven chapters. The LTP should provide any supporting information necessary to address the criteria, including the following:		
4	Chapter 1, General Information	RG 1.179	a. Describe the site characteristics.		
5	Chapter 1, General Information	RG 1.179	b. Identify remaining site dismantlement activities.		
6	Chapter 1, General Information	RG 1.179	c. Discuss plans for site remediation.		
7	Chapter 1, General Information	RG 1.179	d. Provide detailed plans for the final radiation survey for release of the site.		
8	Chapter 1, General Information	RG 1.179	e. Detail a method for demonstrating compliance with the radiological criteria for license termination.		
9	Chapter 1, General Information	RG 1.179	f.(1) Update site-specific estimates of remaining decommissioning costs		

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP (page and paragraph)
10	Chapter 1, General Information	RG 1.179	f.(2) Include the estimated volume of radiological waste and		
11	Chapter 1, General Information	RG 1.179	f.(3) Proposed disposal methods.		
12	Chapter 1, General Information	RG 1.179	g. 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.		
13	Chapter 1, General Information	RG 1.179	h. Identify parts, if any, of the facility that were released for use before approval of the LTP under 10 CFR 50.82(a)(9)(ii)(H).		
14	Chapter 1, General Information	SRP	The LTP is submitted in the form of a supplement to the FSAR or equivalent and the LTP has preceded or is accompanied by an application for license termination.		
15	Chapter 1, General Information	SRP	The LTP is submitted 2 years or more before the proposed termination date of the license.		
16	Chapter 1, General Information	SRP	The LTP is submitted in the form of a license amendment request.		
17	Chapter 1, General Information	SRP	The LTP lists the name and address of the licensee;		
18	Chapter 1, General Information	SRP	license number;		
19	Chapter 1, General Information	SRP	docket number;		
20	Chapter 1, General Information	SRP	facility name and address;		
21	Chapter 1, General Information	SRP	size of the site in acres or square meters;		

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP (page and paragraph)
22	Chapter 1, General Information	SRP	the State and county in which the site is located;		
23	Chapter 1, General Information	SRP	the names of and distances to nearby communities, towns, and cities;		
24	Chapter 1, General Information	SRP	a description of the contours and features of the site;		
25	Chapter 1, General Information	SRP	the elevation of the site;		
26	Chapter 1, General Information	SRP	a description of property surrounding the site, including the location of all off-site wells used by nearby communities or individuals;		
27	Chapter 1, General Information	SRP	the location of the site relative to prominent features such as rivers and lakes;		
28	Chapter 1, General Information	SRP	a map that shows the detailed topography of the site using a contour interval;		
29	Chapter 1, General Information	SRP	the location of the nearest residences and all significant facilities or activities near the site; and		
30	Chapter 1, General Information	SRP	a description of the facilities (buildings, parking lots, fixed equipment, etc.) at the site.		
31	Chapter 1, General Information	SRP	The LTP identifies all changes to the site boundaries (as defined in 10 CFR 20.1003, "Definitions") that have occurred. 10 CFR 50.75(g) requires licensees to keep records that document any changes to the original site boundary such as any partial site release.		
32	Chapter 1, General Information	SRP, App A	Licensee Name and Address:		
33	Chapter 1, General Information	SRP, App A	Docket Number:		

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP (page and paragraph)
34	Chapter 1, General Information	SRP, App A	Facility: name and address of the facility		
35	Chapter 1, General Information	SRP, App A	Facility: location and address of the site		
36	Chapter 1, General Information	SRP, App A	Facility: brief description of the site and immediate environs		
37	Chapter 1, General Information	SRP, App A	Facility: brief description of any changes to the original site boundary [and]		
38	Chapter 1, General Information	SRP, App A	Facility: summary of the licensed activities that occurred at the site		
39	Chapter 1, General Information	SRP, App A	Site Description: size of the site in acres or square meters		
40	Chapter 1, General Information	SRP, App A	Site Description: State and county in which the site is located		
41	Chapter 1, General Information	SRP, App A	Site Description: names and distances to nearby communities, towns and cities		
42	Chapter 1, General Information	SRP, App A	Site Description: description of the contours and features of the site		
43	Chapter 1, General Information	SRP, App A	Site Description: elevation of the site		
44	Chapter 1, General Information	SRP, App A	Site Description: description of property surrounding the site, including the location of all off-site wells used by nearby communities or individuals		
45	Chapter 1, General Information	SRP, App A	Site Description: location of the site relative to prominent features such as rivers and lakes		
46	Chapter 1, General Information	SRP, App A	Site Description: a map that shows the detailed topography of the site using a contour interval		
47	Chapter 1, General Information	SRP, App A	Site Description: the location of the nearest residences and all significant facilities or activities near the site		

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP (page and paragraph)
48	Chapter 1, General Information	SRP, App A	Site Description: description of the facilities (buildings, parking lots, fixed equipment, etc.) at the site and the nature and extent of contamination at the site		
49	Chapter 1, General Information	SRP, App A	Site Description: decommissioning objective proposed by the licensee (i.e., restricted or unrestricted use)		
50	Chapter 2, Site Characterization	RG 1.179	The purpose of the site characterization is to ensure that the licensee conducts final radiation surveys in all areas where contamination existed, remains, or has the potential to exist or remain. NUREG-1575, "Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)," issued August 2000 (Reference 5-1), provides guidance on developing a site characterization program, and NUREG-1757 contains additional guidance.		
51	Chapter 2, Site Characterization	RG 1.179	The licensee can submit the entire site characterization package separately at any time before submitting the LTP and reference it in the LTP, or the licensee can submit the site characterization as an integral part of the LTP.		
52	Chapter 2, Site Characterization	RG 1.179	Extent and range of rad contamination of structures		
53	Chapter 2, Site Characterization	RG 1.179	Extent and range of rad contamination of systems		
54	Chapter 2, Site Characterization	RG 1.179	Extent and range of rad contamination of sewer systems		
55	Chapter 2, Site Characterization	RG 1.179	Extent and range of rad contamination of waste plumbing systems		
56	Chapter 2, Site Characterization	RG 1.179	Extent and range of rad contamination of floor drains		
57	Chapter 2, Site Characterization	RG 1.179	Extent and range of rad contamination of ventilation ducts		

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP (page and paragraph)
58	Chapter 2, Site Characterization	RG 1.179	Extent and range of rad contamination of piping		
59	Chapter 2, Site Characterization	RG 1.179	Extent and range of rad contamination of embedded piping		
60	Chapter 2, Site Characterization	RG 1.179	Extent and range of rad contamination of rubble		
61	Chapter 2, Site Characterization	RG 1.179	Extent and range of rad contamination of paved parking lots - surface		
62	Chapter 2, Site Characterization	RG 1.179	Extent and range of rad contamination of paved parking – buried beneath the site.		
63	Chapter 2, Site Characterization	RG 1.179	Ground water data		
64	Chapter 2, Site Characterization	RG 1.179	Surface water data		
65	Chapter 2, Site Characterization	RG 1.179	Components data		
66	Chapter 2, Site Characterization	RG 1.179	Residues data		
67	Chapter 2, Site Characterization	RG 1.179	Environment data		
68	Chapter 2, Site Characterization	RG 1.179	Maximum contamination levels - structures		
69	Chapter 2, Site Characterization	RG 1.179	Maximum contamination levels - equipment		
70	Chapter 2, Site Characterization	RG 1.179	Maximum contamination levels - soils		
71	Chapter 2, Site Characterization	RG 1.179	Average contamination levels - structures		

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP (page and paragraph)
72	Chapter 2, Site Characterization	RG 1.179	Average contamination levels - equipment		
73	Chapter 2, Site Characterization	RG 1.179	Average contamination levels - soils		
74	Chapter 2, Site Characterization	RG 1.179	Ambient exposure rate measurements - structures		
75	Chapter 2, Site Characterization	RG 1.179	Ambient exposure rate measurements - equipment		
76	Chapter 2, Site Characterization	RG 1.179	Ambient exposure rate measurements - soils		
77	Chapter 2, Site Characterization	RG 1.179	The site characterization should contain sufficiently detailed data to support planning for all remaining decommissioning activities and the final status survey program.		
78	Chapter 2, Site Characterization	RG 1.179	The LTP should describe historic events (including dates, types of occurrences, and locations inside and outside the facility), such as radiological spills, onsite disposals, or other radiological accidents or incidents, that resulted or could have resulted in the contamination of structures, equipment, letdown areas, or soils and ground water beneath buildings and in outside areas.		
79	Chapter 2, Site Characterization	RG 1.179	Describe the survey instruments		
80	Chapter 2, Site Characterization	RG 1.179	Describe the supporting quality assurance (QA) practices used in the site characterization program.		
81	Chapter 2, Site Characterization	RG 1.179	Describe how MARAD applied the data quality objectives discussed in NUREG-1575 during site characterization.		

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP (page and paragraph)
82	Chapter 2, Site Characterization	SRP	The LTP identifies all locations, both inside and outside the facility, where radiological spills, disposals, operational activities, or other radiological accidents and/or incidents occurred and could have resulted in contamination. This identification should be done on a room-by-room or area-by-area basis as necessary, including equipment, laydown areas, or soils (subfloor and outside area).		
83	Chapter 2, Site Characterization	SRP	The LTP describes, in summary form, the original shutdown		
84	Chapter 2, Site Characterization	SRP	The LTP describes, in summary form, the current radiological status of the site		
85	Chapter 2, Site Characterization	SRP	The LTP describes, in summary form, the current non-radiological status of the site		
86	Chapter 2, Site Characterization	SRP	Extent and range of rad contamination of structures		
87	Chapter 2, Site Characterization	SRP	Extent and range of rad contamination of systems		
88	Chapter 2, Site Characterization	SRP	Extent and range of rad contamination of sewer systems		
89	Chapter 2, Site Characterization	SRP	Extent and range of rad contamination of waste management systems,		
90	Chapter 2, Site Characterization	SRP	Extent and range of rad contamination of floor drains		
91	Chapter 2, Site Characterization	SRP	Extent and range of rad contamination of ventilation ducts		
92	Chapter 2, Site Characterization	SRP	Extent and range of rad contamination of piping		
93	Chapter 2, Site Characterization	SRP	Extent and range of rad contamination of embedded piping		

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP (page and paragraph)
94	Chapter 2, Site Characterization	SRP	Extent and range of rad contamination of rubble		
95	Chapter 2, Site Characterization	SRP	Extent and range of rad contamination of ground water		
96	Chapter 2, Site Characterization	SRP	Extent and range of rad contamination of surface water		
97	Chapter 2, Site Characterization	SRP	Extent and range of rad contamination of components		
98	Chapter 2, Site Characterization	SRP	Extent and range of rad contamination of residues		
99	Chapter 2, Site Characterization	SRP	Extent and range of rad contamination of the environment		
100	Chapter 2, Site Characterization	SRP	Maximum contamination levels - structures		
101	Chapter 2, Site Characterization	SRP	Maximum contamination levels - equipment		
102	Chapter 2, Site Characterization	SRP	Maximum contamination levels - soils		
103	Chapter 2, Site Characterization	SRP	Average contamination levels - structures		
104	Chapter 2, Site Characterization	SRP	Average contamination levels - equipment		
105	Chapter 2, Site Characterization	SRP	Average contamination levels - soils		
106	Chapter 2, Site Characterization	SRP	Ambient exposure rate measurements - structures		
107	Chapter 2, Site Characterization	SRP	Ambient exposure rate measurements - equipment		

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP (page and paragraph)
108	Chapter 2, Site Characterization	SRP	Ambient exposure rate measurements - soils		
109	Chapter 2, Site Characterization	SRP	Extent and range of rad contamination of paved parking lots - surface		
110	Chapter 2, Site Characterization	SRP	Extent and range of rad contamination of paved parking – buried beneath the site.		
111	Chapter 2, Site Characterization	SRP	Describe the survey instruments		
112	Chapter 2, Site Characterization	SRP	Describe the supporting quality assurance (QA) practices used in the site characterization program.		
113	Chapter 2, Site Characterization	SRP	Identify the survey instruments used in the site characterization program.		
114	Chapter 2, Site Characterization	SRP	Identify the supporting quality assurance practices used in the site characterization program.		
115	Chapter 2, Site Characterization	SRP	Identify the background levels used during scoping or characterization surveys.		
116	Chapter 2, Site Characterization	SRP	Describe in detail the areas and equipment that need further remediation to allow the reviewer to estimate the radiological conditions that were encountered during remediation of equipment, components, structures, and outdoor areas.		
117	Chapter 2, Site Characterization	SRP, App A	Background Levels Used During Characterization Surveys		
118	Chapter 2, Site Characterization	SRP, App A	Radionuclides Present at Each Location - maximum radionuclide activities (in dpm/100cm2, pCi/gm or pCi/l)		
119	Chapter 2, Site Characterization	SRP, App A	Radionuclides Present at Each Location - average radionuclide activities (in dpm/100cm2, pCi/gm or pCi/l		
120	Chapter 2, Site Characterization	SRP, App A	Radionuclides Present at Each Location - maximum radionuclide ratios, if multiple radionuclides are present (in dpm/100cm2, pCi/gm or pCi/l)		

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP (page and paragraph)
121	Chapter 2, Site Characterization	SRP, App A	Radionuclides Present at Each Location - average radionuclide ratios, if multiple radionuclides are present (in dpm/100cm2, pCi/gm or pCi/l)		
122	Chapter 2, Site Characterization	SRP, App A	Radiological Contamination - List or description of all structures, systems, and equipment at the facility where licensed activities occurred that contain residual radioactive material in excess of site background levels		
123	Chapter 2, Site Characterization	SRP, App A	Radiological Contamination - Summary of the structures, systems, equipment, and locations at the facility that the licensee or responsible party has concluded have not been affected by licensed operations, and the rationale for the conclusion		
124	Chapter 2, Site Characterization	SRP, App A	Radiological Contamination - List or description of each room or area, and equipment within each of the contaminated structures		
125	Chapter 2, Site Characterization	SRP, App A	Radiological Contamination - Summary or map of the locations of contamination in each room or work area		
126	Chapter 2, Site Characterization	SRP, App A	Radiological Contamination - Mode of contamination for each surface (i.e., whether the radioactive material is present only on the surface of the material or if it has penetrated the material)		
127	Chapter 2, Site Characterization	SRP, App A	Characterization Surveys: description and justification of the survey measurements for affected media		
128	Chapter 2, Site Characterization	SRP, App A	Characterization Surveys: survey results, including tables or charts of the concentrations of residual radioactivity measured		
129	Chapter 2, Site Characterization	SRP, App A	Characterization Surveys: maps or drawings of the site, area, or building showing areas classified as impacted or not impacted, with justification for considering areas to be not impacted		

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP (page and paragraph)
130	Chapter 2, Site Characterization	SRP, App A	Surface and Subsurface Soil Contamination: list or description of all locations at the facility where surface and subsurface soil contains residual radioactive material in excess of site background levels		
131	Chapter 2, Site Characterization	SRP, App A	Surface and Subsurface Soil Contamination: scale drawing or map of the site showing the locations of subsurface soil contamination		
132	Chapter 2, Site Characterization	SRP, App A	Surface Water and Ground Water: summary of all surface water bodies and aquifer(s) at the facility that contain residual radioactive material in excess of site background levels		
133	Chapter 2, Site Characterization	NUREG-1757, Vol 2 pg 4-5	The characterization survey provides sufficient information to permit planning for site remediation that would be effective and would not endanger the remediation workers		
134	Chapter 2, Site Characterization	NUREG-1757, Vol 2 pg 4-5	The characterization survey provides sufficient information to demonstrate that it is unlikely that significant quantities of residual radioactivity have gone undetected		
135	Chapter 2, Site Characterization	NUREG-1757, Vol 2 pg 4-5	The characterization survey provides sufficient information to provide information that would be used to design the FSS		
136	Chapter 2, Site Characterization	NUREG-1757, Vol 2, pg 4-10	The characterization survey design is adequate to determine the radiological status of the facility.		
137	Chapter 2, Site Characterization	NUREG-1757, Vol 2, pg 4-10	Describe the radiation characterization survey design		
138	Chapter 2, Site Characterization	NUREG-1757, Vol 2, pg 4-10	Describe the results of the survey, including the following:		

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP (page and paragraph)
139	Chapter 2, Site Characterization	NUREG-1757, Vol 2, pg 4-10	A description and justification of the survey measurements for impacted media (for example, building surfaces, building volumetric, surface soils, subsurface soils, surface water, ground water, sediments, etc., as appropriate)		
140	Chapter 2, Site Characterization	NUREG-1757, Vol 2, pg 4-10	A description of the field instruments that were used for measuring concentrations		
141	Chapter 2, Site Characterization	NUREG-1757, Vol 2, pg 4-10	A description of the methods that were used for measuring concentrations		
142	Chapter 2, Site Characterization	NUREG-1757, Vol 2, pg 4-10	A description of the sensitivities of those field instruments and methods.		
143	Chapter 2, Site Characterization	NUREG-1757, Vol 2, pg 4-10	A description of the laboratory instruments that were used for measuring concentrations.		
144	Chapter 2, Site Characterization	NUREG-1757, Vol 2, pg 4-10	A description of the laboratory methods that were used for measuring concentrations.		
145	Chapter 2, Site Characterization	NUREG-1757, Vol 2, pg 4-10	A description of the sensitivities of those laboratory instruments and methods.		
146	Chapter 2, Site Characterization	NUREG-1757, Vol 2, pg 4-10	The survey results including tables or charts of the concentrations of residual radioactivity measured.		
147	Chapter 2, Site Characterization	NUREG-1757, Vol 2, pg 4-10	Maps or drawings of the site, area, or building showing areas classified as non-impacted or impacted and visually summarizing residual radioactivity concentrations in impacted areas.		

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP (page and paragraph)
148	Chapter 2, Site Characterization	NUREG-1757, Vol 2, pg 4-10	The justification for considering areas to be non-impacted.		
149	Chapter 2, Site Characterization	NUREG-1757, Vol 2, pg 4-10	A discussion of why the licensee considers the characterization survey to be adequate to demonstrate that it is unlikely that significant quantities of residual radioactivity have gone undetected.		
150	Chapter 2, Site Characterization	NUREG-1757, Vol 2, pg 4-10	A discussion of how areas and surfaces were surveyed		
151	Chapter 2, Site Characterization	NUREG-1757, Vol 2, pg 4-10	A discussion of why areas and surfaces did not need to be surveyed - for areas and surfaces that were considered to be inaccessible or not readily accessible		
152	Chapter 2, Site Characterization	NUREG-1757, Vol 2, pg 4-10	For sites, areas, or buildings with multiple radionuclides, a discussion justifying the ratios of radionuclides that were be assumed in the FSS or an indication that no fixed ratio exists and each radionuclide were be measured separately (note that this information may be developed and refined during decommissioning and licensees may elect to include a plan to develop and justify final radionuclide ratios in the LTP).		
153	Chapter 3, Identification of Remaining Site Dismantlement Activities	RG 1.179	Include a discussion of the remaining tasks associated with the decontamination and dismantlement		
154	Chapter 3, Identification of Remaining Site Dismantlement Activities	RG 1.179	Include an estimate of the quantity of radioactive material to be released to unrestricted areas		

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP (page and paragraph)
155	Chapter 3, Identification of Remaining Site Dismantlement Activities	RG 1.179	Include the proposed control mechanisms, dose estimates, and radioactive waste characterization.		
156	Chapter 3, Identification of Remaining Site Dismantlement Activities	RG 1.179	Identify any decommissioning tasks that require coordination with other federal or state regulatory agencies and explain how that coordination occurs.		
157	Chapter 3, Identification of Remaining Site Dismantlement Activities	RG 1.179	Describe the areas and equipment that need further remediation in sufficient detail to allow the reviewer to predict the radiological conditions that were to be encountered during remediation. The details in this section should be sufficient for the NRC to identify any inspection or technical resources needed during the remaining dismantlement activities		
158	Chapter 3, Identification of Remaining Site Dismantlement Activities	RG 1.179	List the remaining activities that do not involve unreviewed safety questions or changes in a facility's technical specifications. This list should be sufficiently detailed for the NRC staff to confirm that remedial activities may in fact be carried out under 10 CFR 50.59		
159	Chapter 3, Identification of Remaining Site Dismantlement Activities	SRP	The LTP discusses the remaining tasks associated with decontamination and dismantlement		
160	Chapter 3, Identification of Remaining Site	SRP	The LTP estimates the quantity of radioactive material to be shipped for disposal or processing		

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP (page and paragraph)
	Dismantlement Activities				
161	Chapter 3, Identification of Remaining Site Dismantlement Activities	SRP	The LTP describes the proposed control mechanisms to ensure that areas are not re-contaminated		
162	Chapter 3, Identification of Remaining Site Dismantlement Activities	SRP	The LTP contains occupational exposure estimates		
163	Chapter 3, Identification of Remaining Site Dismantlement Activities	SRP	The LTP contains radioactive waste characterization		
164	Chapter 3, Identification of Remaining Site Dismantlement Activities	SRP	The LTP describes the remaining dismantlement activities in sufficient detail for the NRC staff to identify any associated inspection or technical resources that were needed.		
165	Chapter 3, Identification of Remaining Site Dismantlement Activities	SRP	The LTP is sufficiently detailed to provide data for use in planning further decommissioning activities.		

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP (page and paragraph)
166	Chapter 3, Identification of Remaining Site Dismantlement Activities	SRP	The LTP includes decontamination techniques.		
167	Chapter 3, Identification of Remaining Site Dismantlement Activities	SRP	The LTP includes projected schedules.		
168	Chapter 3, Identification of Remaining Site Dismantlement Activities	SRP	The LTP includes costs.		
169	Chapter 3, Identification of Remaining Site Dismantlement Activities	SRP	The LTP includes waste volumes.		
170	Chapter 3, Identification of Remaining Site Dismantlement Activities	SRP	The LTP includes dose assessments (including groundwater assessments)		
171	Chapter 3, Identification of Remaining Site Dismantlement Activities	SRP	The LTP includes health and safety considerations.		

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP (page and paragraph)
172	Chapter 3, Identification of Remaining Site Dismantlement Activities	SRP	The LTP lists the remaining activities that do not require any additional licensing action.		
173	Chapter 3, Identification of Remaining Site Dismantlement Activities	SRP, App A	None, See Chapter 4		
174	Chapter 4, Remediation Plans	RG 1.179	Summarize any changes from the previously approved radiological control program that the licensee used for the control of radiological contamination associated with the remaining decommissioning and remediation activities		
175	Chapter 4, Remediation Plans	RG 1.179	Summarize changes to the radiation protection program, but these details should be provided in either periodic updates to the final safety analysis report or the LTP.		
176	Chapter 4, Remediation Plans	RG 1.179	Discuss in detail the remediation methods and techniques that the licensee used to demonstrate that the facility and site areas meet the NRC criteria for license termination in Subpart E of 10 CFR Part 20		
177	Chapter 4, Remediation Plans	RG 1.179	Use of new techniques should be reviewed under the 10 CFR 50.59 criteria and described sufficiently for the NRC to perform a safety evaluation.		
178	Chapter 4, Remediation Plans	SRP	Address any changes in the radiological controls to be implemented to control radiological contamination associated with the remaining decommissioning and remediation activities		

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP (page and paragraph)
179	Chapter 4, Remediation Plans	SRP	Discuss in detail how facility and site areas were to be remediated to meet the proposed residual radioactivity levels (DCGLs) for license termination. Discussions should focus on any unique techniques or procedures used to evaluate whether the DCGLs have been met including the following:		
180	Chapter 4, Remediation Plans	SRP	Summarize the techniques that were used to remediate building structures and components (e.g., scabbling, hydrolazing, grit blasting, etc.).		
181	Chapter 4, Remediation Plans	SRP	Summarize the equipment that was decontaminated and how the decontamination was accomplished.		
182	Chapter 4, Remediation Plans	SRP	Summarize the radiation protection methods and control procedures that was employed including a summary of the procedures already authorized under the existing license.		
183	Chapter 4, Remediation Plans	SRP	Commit to conduct decommissioning activities in accordance with approved written procedures.		
184	Chapter 4, Remediation Plans	SRP	Include a detailed description of the techniques that were employed to remove or remediate surface and subsurface soils, groundwater, and surface water and sediments.		
185	Chapter 4, Remediation Plans	SRP	Describe plans, if any, for onsite disposal of decommissioning waste.		
186	Chapter 4, Remediation Plans	SRP	Include a schedule that demonstrates how and in what time frames MARAD will complete the interrelated decommissioning activities.		
187	Chapter 4, Remediation Plans	SRP, App A	Summary of the radiation protection methods and control procedures that were employed		
188	Chapter 4, Remediation Plans	SRP, App A	Summary of the procedures already authorized under the existing license to conduct decommissioning activities in accordance with approved written procedures		

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP (page and paragraph)
189	Chapter 4, Remediation Plans	SRP, App A	Summary of the procedures for which approval is being requested in the LTP to conduct decommissioning activities in accordance with approved written procedures		
190	Chapter 4, Remediation Plans	SRP, App A	Summary of any unique safety issues associated with remediating contaminated structures		
191	Chapter 4, Remediation Plans	SRP, App A	Summary of any unique safety issues associated with remediating contaminated systems		
192	Chapter 4, Remediation Plans	SRP, App A	Summary of any unique safety issues associated with remediating contaminated equipment		
193	Chapter 4, Remediation Plans	SRP, App A	Summary of any remediation issues associated with remediating contaminated structures		
194	Chapter 4, Remediation Plans	SRP, App A	Summary of any remediation issues associated with remediating contaminated systems		
195	Chapter 4, Remediation Plans	SRP, App A	Summary of any remediation issues associated with remediating contaminated equipment		
196	Chapter 4, Remediation Plans	SRP, App A	Summary of the remediation tasks planned for each room, area and/or system in the order in which they occur		
197	Chapter 4, Remediation Plans	SRP, App A	Description of the remediation techniques that were employed in each room, area, or system		
198	Chapter 4, Remediation Plans	SRP, App A	Summary of the removal and remediation tasks planned for surface and subsurface soil at the site in the order in which they occur, including which activities were conducted by licensee staff and which were performed by a contractor		
199	Chapter 4, Remediation Plans	SRP, App A	Description of the techniques that were employed to remove or remediate surface and subsurface soil at the site		
200	Chapter 4, Remediation Plans	SRP, App A	Summary of the remediation tasks planned for ground and surface water, in the order in which they occur, including which activities were conducted by licensee staff and which were performed by a contractor		

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP (page and paragraph)
201	Chapter 4, Remediation Plans	SRP, App A	Description of the remediation techniques that were employed to remediate the ground or surface water		
202	Chapter 4, Remediation Plans	SRP, App A	Gantt or PERT chart detailing the proposed remediation tasks in the order in which they occur		
203	Chapter 4, Remediation Plans	SRP, App A	Statement acknowledging that circumstances can change during decommissioning		
204	Chapter 4, Remediation Plans	SRP, App A	Statement acknowledging that if the licensee determines that the decommissioning cannot be completed as outlined in the schedule, the MARAD will provide an updated schedule to NRC		
205	Chapter 4, Remediation Plans	NUREG-1757, Vol 2 pg 4-13	The purpose of the review of the description of the remedial action support surveys is to verify that the licensee has designed these surveys appropriately and to assist the licensee in determining when remedial actions have been successful and that the FSS may commence. In addition, information from these surveys may be used to provide the principal estimate of residual radioactivity variability that were used to calculate the FSS sample size in a remediated survey unit.		
206	Chapter 4, Remediation Plans	NUREG-1757, Vol 2 pg 4-13	Describe the remedial action support survey field screening methods.		
207	Chapter 4, Remediation Plans	NUREG-1757, Vol 2 pg 4-13	Describe the remedial action support survey field screening instrumentation.		
208	Chapter 4, Remediation Plans	NUREG-1757, Vol 2 pg 4-13	Demonstration that field screening should be capable of detecting residual radioactivity at the $DCGL_W$.		
209	Chapter 4, Remediation Plans	NUREG-1757, Vol 2 pg 4-13	Describe the remedial action support survey field screening survey instrument sensitivity.		

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP (page and paragraph)
210	Chapter 5, Final Radiation Survey Plan	RG 1.179	Describe the final status survey (FFS). The FSS is the radiation survey performed after an area has been fully characterized and remediated, and MARAD believes that the area is ready to be released. The purpose of the final status survey is to demonstrate that the plant and site meet the radiological criteria for license termination in Subpart E of 10 CFR Part 20.		
211	Chapter 5, Final Radiation Survey Plan	RG 1.179	Describe the methods proposed for surveying all equipment. Use diagrams, plot plans, and facility layout drawings to facilitate presentation.		
212	Chapter 5, Final Radiation Survey Plan	RG 1.179	Describe the methods proposed for surveying all systems. Use diagrams, plot plans, and facility layout drawings to facilitate presentation.		
213	Chapter 5, Final Radiation Survey Plan	RG 1.179	Describe the methods proposed for surveying all structures. Use diagrams, plot plans, and facility layout drawings to facilitate presentation.		
214	Chapter 5, Final Radiation Survey Plan	RG 1.179	Describe the methods proposed for surveying all soils. Use diagrams, plot plans, and facility layout drawings to facilitate presentation.		
215	Chapter 5, Final Radiation Survey Plan	RG 1.179	Describe the method for ensuring that sufficient data are included for a meaningful statistical survey.		
216	Chapter 5, Final Radiation Survey Plan	RG 1.179	Describe the methods the licensee used to establish background radiation levels.		
217	Chapter 5, Final Radiation Survey Plan	RG 1.179	Discuss variances in background radiation that can be expected		
218	Chapter 5, Final Radiation Survey Plan	RG 1.179	Describe the QA program to support field survey work		

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP (page and paragraph)
219	Chapter 5, Final Radiation Survey Plan	RG 1.179	Describe the QA program to support laboratory analysis		
220	Chapter 5, Final Radiation Survey Plan	RG 1.179	Describe the QA organization		
221	Chapter 5, Final Radiation Survey Plan	RG 1.179	Describe the QA program for training and qualification requirements		
222	Chapter 5, Final Radiation Survey Plan	RG 1.179	Describe the QA program for survey instructions and procedures, including water, air, and soil sampling procedures		
223	Chapter 5, Final Radiation Survey Plan	RG 1.179	Describe the QA program for document control		
224	Chapter 5, Final Radiation Survey Plan	RG 1.179	Describe the QA program for control of purchased items		
225	Chapter 5, Final Radiation Survey Plan	RG 1.179	Describe the QA program for inspections		
226	Chapter 5, Final Radiation Survey Plan	RG 1.179	Describe the QA program for control of survey equipment - handling		
227	Chapter 5, Final Radiation Survey Plan	RG 1.179	Describe the QA program for control of survey equipment - storage		
228	Chapter 5, Final Radiation Survey Plan	RG 1.179	Describe the QA program for control of survey equipment - response checks		

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP (page and paragraph)
229	Chapter 5, Final Radiation Survey Plan	RG 1.179	Describe the QA program for shipping of survey equipment		
230	Chapter 5, Final Radiation Survey Plan	RG 1.179	Describe the QA program for shipping of survey laboratory samples		
231	Chapter 5, Final Radiation Survey Plan	RG 1.179	Describe the QA program for disposition of nonconformance items		
232	Chapter 5, Final Radiation Survey Plan	RG 1.179	Describe the QA program for corrective action		
233	Chapter 5, Final Radiation Survey Plan	RG 1.179	Describe the QA program for QA records		
234	Chapter 5, Final Radiation Survey Plan	RG 1.179	Describe the QA program for survey audits, including methods to be used for reviewing, analyzing, and auditing data.		
235	Chapter 5, Final Radiation Survey Plan	RG 1.179	Describe the verification surveys and evaluations used to support the delineation of radiologically affected (contaminated) areas		
236	Chapter 5, Final Radiation Survey Plan	RG 1.179	Describe the verification surveys and evaluations used to support the delineation of unaffected (uncontaminated) areas		
237	Chapter 5, Final Radiation Survey Plan	RG 1.179	Identify the major radiological contaminants.		
238	Chapter 5, Final Radiation Survey Plan	RG 1.179	Discuss methods used for addressing hard-to-detect radionuclides.		

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP (page and paragraph)
239	Chapter 5, Final Radiation Survey Plan	RG 1.179	Describe access control procedures to avoid recontamination of clean areas.		
240	Chapter 5, Final Radiation Survey Plan	RG 1.179	Identify survey units having the same area classification.		
241	Chapter 5, Final Radiation Survey Plan	RG 1.179	Describe scanning performed to locate small areas of elevated concentrations of residual radioactivity.		
242	Chapter 5, Final Radiation Survey Plan	RG 1.179	Discuss levels established for investigating significantly elevated concentrations of residual radioactivity. Include survey instrument calibration and efficiency calculations.		
243	Chapter 5, Final Radiation Survey Plan	RG 1.179	Describe the reference coordinate system established for the site areas.		
244	Chapter 5, Final Radiation Survey Plan	SRP	Identify the major radiological contaminants		
245	Chapter 5, Final Radiation Survey Plan	SRP	Methods used for addressing hard-to-detect radionuclides		
246	Chapter 5, Final Radiation Survey Plan	SRP	Access control procedures to control recontamination of clean areas		
247	Chapter 5, Final Radiation Survey Plan	SRP	Describe the QA program to support field survey work		
248	Chapter 5, Final Radiation Survey Plan	SRP	Describe the QA program to support laboratory analysis		

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP (page and paragraph)
249	Chapter 5, Final Radiation Survey Plan	SRP	Describe the QA organization		
250	Chapter 5, Final Radiation Survey Plan	SRP	Describe the QA program for training and qualification requirements		
251	Chapter 5, Final Radiation Survey Plan	SRP	Describe the QA program for survey instructions and procedures, including water, air, and soil sampling procedures		
252	Chapter 5, Final Radiation Survey Plan	SRP	Describe the QA program for document control		
253	Chapter 5, Final Radiation Survey Plan	SRP	Describe the QA program for control of purchased items		
254	Chapter 5, Final Radiation Survey Plan	SRP	Describe the QA program for inspections		
255	Chapter 5, Final Radiation Survey Plan	SRP	Describe the QA program for control of survey equipment - handling		
256	Chapter 5, Final Radiation Survey Plan	SRP	Describe the QA program for control of survey equipment - storage		
257	Chapter 5, Final Radiation Survey Plan	SRP	Describe the QA program for control of survey equipment - calibration (NOT in RG 1.179)		
258	Chapter 5, Final Radiation Survey Plan	SRP	Describe the QA program for control of survey equipment - response checks		

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP (page and paragraph)
259	Chapter 5, Final Radiation Survey Plan	SRP	Describe the QA program for shipping of survey equipment		
260	Chapter 5, Final Radiation Survey Plan	SRP	Describe the QA program for shipping of survey laboratory samples		
261	Chapter 5, Final Radiation Survey Plan	SRP	Describe the QA program for nonconformance items		
262	Chapter 5, Final Radiation Survey Plan	SRP	Describe the QA program for corrective action		
263	Chapter 5, Final Radiation Survey Plan	SRP	Describe the QA program for QA records		
264	Chapter 5, Final Radiation Survey Plan	SRP	Describe the QA program for survey audits, including methods to be used for reviewing, analyzing, and auditing data.		
265	Chapter 5, Final Radiation Survey Plan	SRP	Methods for surveying embedded and buried piping		
266	Chapter 5, Final Radiation Survey Plan	SRP	Final survey plan meets the evaluation criteria defined in Section 4 of NUREG-1757, Vol. 2. Included below lines in 335 to 349		
267	Chapter 5, Final Radiation Survey Plan	SRP App A	Summary table or list of the $DCGL_W$ for each radionuclide and affected media of concern		
268	Chapter 5, Final Radiation Survey Plan	SRP App A	If Class 1 survey units are present, a summary table or list of area factors that were be used to determine the $DCGL_{EMC}$ for each radionuclide and media of concern		

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP (page and paragraph)
269	Chapter 5, Final Radiation Survey Plan	SRP App A	If Class 1 survey units are present, the $DCGL_{EMC}$ for each radionuclide and medium of concern		
270	Chapter 5, Final Radiation Survey Plan	SRP App A	If multiple radionuclides are present, the appropriate $DCGL_{W}$ for the survey method to be used		
271	Chapter 5, Final Radiation Survey Plan	SRP App A	Discussion of why the licensee considers the characterization survey to be adequate to demonstrate that it is unlikely that significant quantities of residual radioactivity have gone undetected		
272	Chapter 5, Final Radiation Survey Plan	SRP App A	For areas and surfaces that are inaccessible or not readily accessible, a discussion of how they were surveyed or why they did not need to be surveyed		
273	Chapter 5, Final Radiation Survey Plan	SRP App A	For sites, areas, or buildings with multiple radionuclides, a discussion justifying the ratios of radionuclides that were assumed in the final status survey or an indication that no fixed ratio exists and each radionuclide was measured separately		
274	Chapter 5, Final Radiation Survey Plan	SRP App A	Remediation Survey: description of field screening methods and instrumentation		
275	Chapter 5, Final Radiation Survey Plan	SRP App A	Remediation Survey: demonstration that field screening should be capable of detecting residual radioactivity at 10-50 percent of the DCGL		
276	Chapter 5, Final Radiation Survey Plan	SRP App A	Final Status Survey Design: brief overview describing the final status survey design		
277	Chapter 5, Final Radiation Survey Plan	SRP App A	Final Status Survey Design: description and map or drawing of affected areas of the site, area, or buildings classified by residual radioactivity levels (Class 1, Class 2, or Class 3) and divided into		

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP (page and paragraph)
			survey units with an explanation of the basis for division into survey units		
278	Chapter 5, Final Radiation Survey Plan	SRP App A	Final Status Survey Design: description of the background reference areas and materials, if they were used, and a justification for their selection		
279	Chapter 5, Final Radiation Survey Plan	SRP App A	Final Status Survey Design: summary of the statistical tests that were used to evaluate the survey results • description of scanning instruments, methods, calibration, operational checks, coverage, and sensitivity for each media and radionuclide		
280	Chapter 5, Final Radiation Survey Plan	SRP App A	Final Status Survey Design: for in situ sample measurements made by field instruments, a description of the instruments, calibration, operational checks, sensitivity, and sampling methods with a demonstration that the instruments and methods have adequate sensitivity		
281	Chapter 5, Final Radiation Survey Plan	SRP App A	Final Status Survey Design: description of the analytical instruments for measuring samples in the laboratory, including their calibration, sensitivity, and methods with a demonstration that the instruments have adequate sensitivity		
282	Chapter 5, Final Radiation Survey Plan	SRP App A	Final Status Survey Design: description of how the samples to be analyzed in the laboratory were collected, controlled, and handled		
283	Chapter 5, Final Radiation Survey Plan	SRP App A	Final Status Survey Design: description of the final status survey investigation levels and how they were determined		
284	Chapter 5, Final Radiation Survey Plan	SRP App A	Final Status Survey Design: summary of any significant additional residual radioactivity that was not accounted for during site characterization		

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285	Chapter 5, Final Radiation Survey Plan	SRP App A	Final Status Survey Design: summary of direct measurement results and/or soil concentration levels in units that are comparable to the DCGL, and whether data are used to estimate or update the survey unit		
286	Chapter 5, Final Radiation Survey Plan	SRP App A	Final Status Survey Design: description of performance of confirmatory surveys		
287	Chapter 5, Final Radiation Survey Plan	SRP App A	Final Status Survey Design: description of performance of split sampling		
288	Chapter 5, Final Radiation Survey Plan	SRP App A	Final Status Survey Design: description of performance of side by side measurements		
289	Chapter 5, Final Radiation Survey Plan	SRP App A	Final Status Survey Design: summary of the direct measurements or sample data used to evaluate the success of remediation and estimate the survey unit variance		
290	Chapter 5, Final Radiation Survey Plan	SRP App A	Quality Assurance Program to Support Final Surveys: description of the QA program management organization, the duties and responsibilities of each unit within the organization, how delegation of responsibilities is managed within the decommissioning program, and how work performance is evaluated		
291	Chapter 5, Final Radiation Survey Plan	SRP App A	Quality Assurance Program to Support Final Surveys: description of the authority of each unit within the QA program		
292	Chapter 5, Final Radiation Survey Plan	SRP App A	Quality Assurance Program to Support Final Surveys: organization chart of the QA program		

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP (page and paragraph)
293	Chapter 5, Final Radiation Survey Plan	SRP App A	Quality Assurance Program to Support Final Surveys: commitment that activities affecting the quality of site decommissioning were subject to the applicable controls of the QA program, and activities covered by the QA program are identified in program-defining documents		
294	Chapter 5, Final Radiation Survey Plan	SRP App A	Quality Assurance Program to Support Final Surveys: description of the self-assessment program to confirm that activities affecting quality comply with the QA program		
295	Chapter 5, Final Radiation Survey Plan	SRP App A	Quality Assurance Program to Support Final Surveys: commitment that persons performing self-assessment activities do not have direct responsibilities in the area they assess		
296	Chapter 5, Final Radiation Survey Plan	SRP App A	Final Status Survey Report: overview of the results of the final status survey		
297	Chapter 5, Final Radiation Survey Plan	SRP App A	Final Status Survey Report: discussion of any changes that were made in the final status survey from what was proposed in the LTP		
298	Chapter 5, Final Radiation Survey Plan	SRP App A	Final Status Survey Report: description of the method by which the number of samples was determined for each survey unit		
299	Chapter 5, Final Radiation Survey Plan	SRP App A	Final Status Survey Report: summary of the values used to determine the number of samples and a justification for these values		
300	Chapter 5, Final Radiation Survey Plan	SRP App A	Final Status Survey Report: survey results for each survey unit including the number of samples taken for the survey unit, and a map or drawing of the survey unit showing the reference system and random start systematic sample locations for Class 1 and Class 2 survey units and random locations for Class 3 survey units and reference areas		

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP (page and paragraph)
301	Chapter 5, Final Radiation Survey Plan	SRP App A	Final Status Survey Report: measured sample concentrations		
302	Chapter 5, Final Radiation Survey Plan	SRP App A	Final Status Survey Report: statistical evaluation of the measured concentrations, survey instrument calibration procedures, and survey instrument efficiency calculations		
303	Chapter 5, Final Radiation Survey Plan	SRP App A	Final Status Survey Report: Final Status Survey Report: judgmental and miscellaneous sample data sets, reported separately from those samples collected for performing the statistical evaluation		
304	Chapter 5, Final Radiation Survey Plan	SRP App A	Final Status Survey Report: discussion of anomalous data, including any areas of elevated direct radiation detected during scanning that exceeded the investigation level or measurement locations in excess of $DCGL_W$		
305	Chapter 5, Final Radiation Survey Plan	SRP App A	Final Status Survey Report: statement that a given survey unit satisfied the $DCGL_W$ and the elevated measurement comparison if any sample points exceeded the $DCGL_W$		
306	Chapter 5, Final Radiation Survey Plan	SRP App A	Final Status Survey Report: if survey unit fails, description of any changes in initial survey unit assumptions relative to the extent of residual radioactivity, the investigation conducted to ascertain the reason for the failure and the effect that the failure has on the conclusion that the facility is ready for final radiological surveys; and if a survey unit fails, a discussion of the effect of the failure has on other survey unit information		
307	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Rev. 1, Vol. 2 Pg 4-5	NRC staff should review the FSS design to determine whether the survey design is adequate for demonstrating compliance with the radiological criteria for license termination.		

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308	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Rev. 1, Vol. 2 Pg 4-5	NRC staff should review the results of the FSS to determine whether the survey demonstrates that the site, area, or building meets the radiological criteria for license termination.		
309	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Rev. 1, Vol. 2 Pg 4-5	NRC staff should note that NRC regulations require that LTPs include a description of the planned final radiological survey. Recognizing the flexible approach discussed in Section 2.2 of NUREG 1757, Vol 2 and that the MARSSIM approach allows certain information needed to develop the final radiological survey to be obtained as part of the remedial activities at the site, a licensee or responsible party may submit information on facility radiation surveys in one of two ways, as summarized below. Section 2.2 of NUREG 1757, Vol 2 provides additional relevant guidance.		
310	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Rev. 1, Vol. 2 Pg 4-6	MARAD should list the DCGL(s) that were be used to design the surveys and to demonstrate compliance with the radiological criteria for release, including … (next 4 lines)		
311	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Rev. 1, Vol. 2 Pg 4-6	Include a summary table or list of the $DCGL_W$ for each radionuclide and impacted medium of concern		
312	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Rev. 1, Vol. 2 Pg 4-6	Include a summary table or list of area factors that were be used for determining a $DCGL_{EMC}$ for each radionuclide and media of concern if Class 1 (refer to Appendix A.1 of this volume for classification of site areas) survey units are present		
313	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Rev. 1, Vol. 2 Pg 4-6	Include the $DCGL_{EMC}$ for each radionuclide and medium of concern if Class 1 survey units are present		
314	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Rev. 1, Vol. 2 Pg 4-6	Include the appropriate $DCGL_W$ for the survey method to be used if multiple radionuclides are present.		

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP (page and paragraph)
315	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Rev. 1, Vol. 2 Pg 4-6	NRC staff should verify that, for each radionuclide and impacted media of concern, MARAD has provided a $DCGL_W$ and, if Class 1 survey units are present, a table of area factors.		
316	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Rev. 1, Vol. 2 Pg 4-6	NRC staff should verify that the values presented are consistent with the values developed pursuant to the dose modeling, as discussed in Chapter 5 of NUREG 1715, Vol 2.		
317	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Rev. 1, Vol. 2 Pg 4-6	If multiple radionuclides are present, MARSSIM Sections 4.3.2, 4.3.3, and 4.3.4 of NUREG 1757, Vol 2 describe acceptable methods to determine DCGLs appropriate for the survey technique		
318	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Rev. 1, Vol. 2 Pg 4-14	FSS primary object 1 of 3: verify survey unit classification		
319	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Rev. 1, Vol. 2 Pg 4-14	FSS primary object 2 of 3: demonstrate that the potential dose from residual radioactivity is below the release criterion for each survey unit		
320	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Rev. 1, Vol. 2 Pg 4-14	FSS primary object 3 of 3: demonstrate that the potential dose from small areas of elevated activity is below the release criterion for each survey unit.		
321	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Rev. 1, Vol. 2 Pg 4-14	Data provided by the FSS can demonstrate that all radiological parameters satisfy the established guideline values and conditions		
322	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Rev. 1, Vol. 2 Pg 4-14	The purpose of NRC staff's review is to verify that the design of the FSS is adequate to demonstrate compliance with the radiological criteria for license termination		

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP (page and paragraph)
323	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Rev. 1, Vol. 2 Pg 4-14	The information supplied by MARAD should be sufficient to allow NRC staff to determine that the FSS design is adequate to demonstrate compliance with the radiological criteria for license termination.		
324	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Rev. 1, Vol. 2 Pg 4-14	FSS design includes a brief overview describing the FSS design		
325	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Rev. 1, Vol. 2 Pg 4-14	FSS design includes a description and map or drawing of impacted areas of the site, area, or building classified by residual radioactivity levels (Class 1, 2, or 3) and divided into survey units, with an explanation of the basis for division into survey units (maps should have compass headings indicated)		
326	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Rev. 1, Vol. 2 Pg 4-14	FSS design includes a description of the background reference areas and materials, if they were used, and a justification for their selection		
327	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Rev. 1, Vol. 2 Pg 4-14	FSS design includes a summary of the statistical tests that were used to evaluate the survey results, including the elevated measurement comparison, if Class 1 survey units are present; a justification for any test methods not included in MARSSIM; and the values for the decision errors (α and β) with a justification for α values greater than 0.05		
328	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Rev. 1, Vol. 2 Pg 4-14	FSS design includes a description of scanning instruments, methods, calibration, operational checks, coverage, and sensitivity for each media and radionuclide		

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP (page and paragraph)
329	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Rev. 1, Vol. 2 Pg 4-14	FSS design includes a description of the instruments, calibration, operational checks, sensitivity, and sampling methods for in situ sample measurements, with a demonstration that the instruments and methods have adequate sensitivity		
330	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Rev. 1, Vol. 2 Pg 4-14	FSS design includes a description of the analytical instruments for measuring samples in the laboratory, including the calibration, sensitivity, and methodology for evaluation, with a demonstration that the instruments and methods have adequate sensitivity		
331	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Rev. 1, Vol. 2 Pg 4-14	FSS design includes a description of how the samples to be analyzed in the laboratory were collected, controlled, and handled		
332	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Rev. 1, Vol. 2 Pg 4-14	FSS design includes a description of the FSS investigation levels and how they were determined.		
333	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Rev. 1, Vol. 2 Pg 4-15	Evaluation Criteria: Appendix A of NUREG 1757 Vol. 2, for general guidance on implementing the MARSSIM approach for conducting FSSs		
334	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Rev. 1, Vol. 2 Pg 4-15	Evaluation Criteria: Appendix B of NUREG 1757 Vol. 2, for guidance on alternative methods of FSS for simple situations		
335	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Rev. 1, Vol. 2 Pg 4-15	Evaluation Criteria: MARSSIM Sections 4.4 and 4.6 for classifying areas by residual radioactivity levels and dividing areas into survey units of acceptable size		

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP (page and paragraph)
336	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Rev. 1, Vol. 2 Pg 4-15	Evaluation Criteria: MARSSIM Section 4.5 for methods to select background reference areas and materials		
337	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Rev. 1, Vol. 2 Pg 4-15	Evaluation Criteria: NUREG–1505, Chapter 13, for a method to account for differences in background concentrations between different reference areas		
338	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Rev. 1, Vol. 2 Pg 4-15	Evaluation Criteria: MARSSIM Section 5.5.2 for statistical tests		
339	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Rev. 1, Vol. 2 Pg 4-15	Evaluation Criteria: Appendix A of NUREG 1757 Vol. 2, Section A.7.2 for decision errors		
340	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Rev. 1, Vol. 2 Pg 4-15	Evaluation Criteria: MARSSIM Sections 6.5.3 and 6.5.4 for selection of acceptable survey instruments, calibration, and operational checkout methods		
341	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Rev. 1, Vol. 2 Pg 4-15	Evaluation Criteria: MARSSIM Section 6.7 for methods to determine measurement sensitivity		
342	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Rev. 1, Vol. 2 Pg 4-15	Evaluation Criteria: NUREG–1507 for instrument sensitivity information		

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP (page and paragraph)
343	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Rev. 1, Vol. 2 Pg 4-15	Evaluation Criteria: MARSSIM Sections 5.5.2.4, 5.5.2.5, 5.5.3, 7.5, and 7.6 for scanning and sampling		
344	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Rev. 1, Vol. 2 Pg 4-15	Evaluation Criteria: MARSSIM Section 7.7 for sample analytical methods (Table 7.2 of Section 7.7 provides acceptable analytical procedural references)		
345	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Rev. 1, Vol. 2 Pg 4-15	Evaluation Criteria: MARSSIM Sections 7.5 and 7.6 for methods for sample collection		
346	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Rev. 1, Vol. 2 Pg 4-15	Evaluation Criteria: MARSSIM Section 5.5.2.6 for survey investigation levels		
347	Chapter 5, Final Radiation Survey Plan	NUREG-1757, Rev. 1, Vol. 2 Pg 4-15	Evaluation Criteria: Appendix G of NUREG 1757 Vol. 2 for surveys for special structural or land situations.		
348	Chapter 6, Dose Modeling	RG 1.179	The LTP should demonstrate that the dose from residual radioactivity that is distinguishable from background radiation is less than the limits in Subpart E of 10 CFR Part 20.		
349	Chapter 6, Dose Modeling	RG 1.179	The LTP should also demonstrate that residual radioactivity has been reduced to levels that are as low as reasonably achievable (ALARA) (see 10 CFR 20.1402, “Radiological Criteria for Unrestricted Use”).		
350	Chapter 6, Dose Modeling	RG 1.179	The LTP should describe in detail the methods and assumptions used to demonstrate compliance with the 25-mrem (0.25-mSv)-per-year criterion.		

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP (page and paragraph)
351	Chapter 6, Dose Modeling	RG 1.179	NUREG-1757 Vol 2, Sec 5 and its Appendix H provide additional guidance on how to demonstrate compliance with the unrestricted release. (See lines 365 to 390 below.)		
352	Chapter 6, Dose Modeling	SRP	If MARAD desires an unrestricted release in accordance with the requirements of 10 CFR 20.1402, "Radiological Criteria for Unrestricted Use," the LTP should describe the methods used to demonstrate compliance.		
353	Chapter 6, Dose Modeling	SRP	The information that should be submitted in the LTP and the associated evaluation criteria are described in NUREG-1757. Group 4-5		
354	Chapter 6, Dose Modeling	SRP	NUREG 1757, Vol 2, Group 4-5 Unrestricted release using screening criteria - 5.1 and Appendix H (Group 4-5) (See lines 365 to 390 below.)		
355	Chapter 6, Dose Modeling	SRP	Deleted		
356	Chapter 6, Dose Modeling	SRP	NUREG 1757, Vol 2 As Low As Is Reasonably Achievable (ALARA) 6.0 and Appendix N (See lines 365 to 390 below.)		
357	Chapter 6, Dose Modeling	SRP App A	For unrestricted release using screening criteria for building surface residual radioactivity, the general conceptual model (for both the source term and the building environment) of the site, and a summary of the screening method.		
358	Chapter 6, Dose Modeling	SRP App A	For unrestricted release using screening criteria for surface soil residual radioactivity, justification on the appropriateness of using the screening approach (for both the source term and the environment) at the site, as well as a summary of the screening method (i.e., running DandD or using the lookup tables).		

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP (page and paragraph)
359	Chapter 6, Dose Modeling	SRP App A	ALARA Analysis: description of how the licensee or responsible party should demonstrate that residual radioactivity levels are ALARA		
360	Chapter 6, Dose Modeling	SRP App A	ALARA Analysis: quantitative cost-benefit analysis		
361	Chapter 6, Dose Modeling	SRP App A	ALARA Analysis: description of how costs were estimated		
362	Chapter 6, Dose Modeling	SRP App A	ALARA Analysis: a demonstration that the doses to the average member of the critical group are ALARA		
363	Chapter 6, Dose Modeling	NUREG-1757, Rev. 1, Vol. 2 Pg 5-1	NRC staff review of dose modeling consists of evaluations of four general areas: 1. the source term assumptions, 2. an exposure scenario considering the site environment, 3. the mathematical model/computational method used, and 4. the parameter values and a measure of their uncertainty.		
364	Chapter 6, Dose Modeling	NUREG-1757, Rev. 1, Vol. 2 Pg 5-10	Dose modeling for building surfaces using the default screening criteria should include both of the following: 1) the general conceptual model (for both the source term and the building or outside environment) of the site, and 2) a summary of the screening method (i.e., running DandD or using the look-up tables) used.		
365	Chapter 6, Dose Modeling	NUREG-1757, Rev. 1, Vol. 2 Pg 5-11	If MARAD uses the default screening methods and parameters inherent in the DandD code by either running the computer code or using look-up tables, the reviewers should only need to review: 1) the source term model and 2) the overall applicability of using the screening method with the associated residual radioactivity.		

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP (page and paragraph)
366	Chapter 6, Dose Modeling	NUREG-1757, Rev. 1, Vol. 2 Pg 5-11	NRC staff determined the acceptability of the MARAD projections of (a) radiological impacts on future individuals from residual radioactivity and (b) compliance with regulatory criteria. The information may be considered acceptable if it is sufficient to ensure a reasonable assessment of the possible future impacts from the residual radioactivity on building surfaces.		
367	Chapter 6, Dose Modeling	NUREG-1757, Rev. 1, Vol. 2 Sec 5.1.1	Source Term Configuration Confirm that the actual measurements, facility history, and planned remedial action(s) support the source term configuration used in the modeling by reviewing the information in the facility history, radiological status, and planned remedial action(s) portions of the LTP.		
368	Chapter 6, Dose Modeling	NUREG-1757, Rev. 1, Vol. 2 Sec 5.1.1	Source Term Configuration Determine if the physical configuration of the residual radioactivity can adequately be assumed to be a thin layer of residual radioactivity on the building surfaces		
369	Chapter 6, Dose Modeling	NUREG-1757, Rev. 1, Vol. 2 Sec 5.1.1	Source Term Configuration IF the residual radioactivity is not limited to the building surfaces, THEN 1) use of the default screening criteria are not warranted without additional justification AND 2) the NRC reviewer should reclassify MARAD as a Group 4 licensee and evaluate the modeling using Section 5.2. NOTE: Sec 5.2 should not be added to the matrix unless it is needed.]		

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP (page and paragraph)
370	Chapter 6, Dose Modeling	NUREG-1757, Rev. 1, Vol. 2 Sec 5.1.1	Residual Radioactivity Spatial Variability Based on NSS conditions both before and those projected after the decommissioning alternative, confirm it is appropriate to make an assumption of homogeneity (a) for the whole facility OR (b) for subsections of the facility.		
371	Chapter 6, Dose Modeling	NUREG-1757, Rev. 1, Vol. 2 Sec 5.1.1	Residual Radioactivity Spatial Variability Confirm the adequacy of the determination of a representative value (or range of values) for the residual radioactivity concentration in the source term modeled.		
372	Chapter 6, Dose Modeling	NUREG-1757, Rev. 1, Vol. 2 Sec 5.1.1	Execution of the DandD Computer Code Dose Calculations Verify the residual radioactivity is limited to building surfaces.		
373	Chapter 6, Dose Modeling	NUREG-1757, Rev. 1, Vol. 2 Sec 5.1.1	Execution of the DandD Computer Code Dose Calculations If the appropriate annual peak dose is greater than 0.025 mSv (2.5 mrem), verify the removable fraction of the residual radioactivity is 10 % or less at the time of license termination, or verify the removable fraction has been adjusted as explained in footnote A in Table H.1.		
374	Chapter 6, Dose Modeling	NUREG-1757, Rev. 1, Vol. 2 Sec 5.1.1	Execution of the DandD Computer Code Dose Calculations Verify the output reports indicate that no parameters (other than source term concentrations) were modified.		
375	Chapter 6, Dose Modeling	NUREG-1757, Rev. 1, Vol. 2 Sec 5.1.1	Execution of the DandD Computer Code Dose Calculations Verify use of the 90th percentile of the dose distribution to compare with the dose limit		

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP (page and paragraph)
376	Chapter 6, Dose Modeling	NUREG-1757, Rev. 1, Vol. 2 Sec 5.1.1	DCGLs from the DandD Code or Look-up Tables Verify use of either the DandD computer code or the published look-up table for beta and gamma emitters (see Appendix H) to establish radionuclide-specific DCGLs equivalent to 0.25 mSv/y (25 mrem/y).		
377	Chapter 6, Dose Modeling	NUREG-1757, Rev. 1, Vol. 2 Sec 5.1.1	DCGLs from the DandD Code or Look-up Tables If using radionuclide-specific DCGLs, verify that each of the following three conditions are true: 1. The residual radioactivity is limited to building surfaces. 2. If the residual radioactivity is greater than 10 % of the respective screening DCGLs (Table H.1 from Appendix H of this volume [NUREG-1757, Vol. 2, Rev. 1]), the removable fraction is 10 % or less at license termination, or the removable fraction has been adjusted as explained in footnote A in Table H.1. 3. If more than one radionuclide is involved, there is reasonable assurance that the sum of fractions (concentrations divided by DCGLs) is no greater than 1. See NUREG-1757, Vol. 2, Section 2.7.		
378	Chapter 6, Dose Modeling	NUREG-1757, Rev. 1, Vol. 2 Sec 5.1.1	DCGLs from the DandD Code or Look-up Tables If using the DandD Version 2 computer code to calculate the radionuclide-specific DCGLs, verify that each of the following two conditions are true: 1. The output reports indicate that no parameters (other than entering unit concentrations) were modified. 2. The 90th percentile of the dose distribution were used to derive the concentrations.		

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP (page and paragraph)
379	Chapter 6, Dose Modeling	NUREG-1757, Rev. 1, Vol. 2 Sec 5.1.1	C. Compliance with Regulatory Criteria Confirm projections of compliance with regulatory criteria, if that decommissioning option is pursued, are acceptable provided that NRC staff has reasonable assurance that at least one of the following is true: 1. The only residual radioactivity is on building surfaces, and the level of removable residual radioactivity does not violate the assumptions in the model. 2. The final concentrations result in a peak annual dose of less than 0.25 mSv (25 mrem) and the licensee has committed to calculating the annual dose using a screening analysis at license termination. 3. The planned DCGLs are equal to or less than those provided by the screening criteria, and the licensee has committed to maintaining the sum of fractions, if applicable.		
380	Chapter 6, Dose Modeling	NUREG-1757, Vol. 2, Rev. 1, Appendix H.1	Confirm use of NRC staff review of the screening analysis should be performed using one or more of the currently available screening tools: 1. A look-up table for common beta- and gamma-emitting radionuclides for building surface residual radioactivity (63 FR 64132, November 18, 1998); and 2. Screening levels derived using DandD, Version 2.1, or the most current version for the specific radionuclide(s) that use the code's default parameters.		
381	Chapter 6, Dose Modeling	NUREG-1757, Vol. 2, Rev. 1, Appendix H.2	Confirm use of the of either a) NRC's look-up tables in 63 FR 64132 and 64 FR 68395, or (b) the latest version (e.g., Version 2.1) of the DandD code developed by NRC to perform the generic screening analysis.		

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP (page and paragraph)
382	Chapter 6, Dose Modeling	NUREG-1757, Vol. 2, Rev. 1, Appendix H.2	When using the screening approach for demonstrating compliance with the dose criteria in Part 20, Subpart E, confirm that the particular site conditions (e.g., physical and source-term conditions) are compatible and consistent with the DandD model assumptions.		
383	Chapter 6, Dose Modeling	NUREG-1757, Vol. 2, Rev. 1, Appendix H.2	Confirm the default parameters and default scenarios and pathways are used in the screening dose analysis.		
384	Chapter 6, Dose Modeling	NUREG-1757, Vol. 2, Rev. 1, Appendix H.2	Confirm the site is qualified for screening by examining the site conceptual model, the generic source-term characteristics, and other attributes of the site.		
385	Chapter 6, Dose Modeling	NUREG-1757, Vol. 2, Rev. 1, Appendix H	1. Confirm the residual radioactivity on building surfaces (e.g., walls, floors, ceilings) should be surficial and non-volumetric [e.g., #10 mm (0.39 in) of penetration].		
386	Chapter 6, Dose Modeling	NUREG-1757, Vol. 2, Rev. 1, Appendix H	2. Confirm residual radioactivity on building surfaces is mostly fixed (not loose), with the fraction of loose (removable) residual radioactivity no greater than 10 % of the total surface activity. Note that for cases when the fraction of removable contamination is undetermined or higher than 0.1, licensees may assume for screening purposes that 100 % of surface contamination is removable, and therefore the screening values should be decreased by a factor of 10 (see footnote A to Table H.1).		

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP (page and paragraph)
387	Chapter 6, Dose Modeling	NUREG-1757, Vol. 2, Rev. 1, Appendix H	3. Confirm the screening criteria are not being applied to building surfaces such as buried structures (e.g., drainage or sewer pipes) or equipment within the building without adequate justification; such structures, buried surfaces, and clearance of equipment should be treated on a case-by-case basis.		
388	Chapter 6, Dose Modeling	NUREG-1757, Vol. 2, Rev. 1, Appendix H	Verify use of look-up table (Table H.1) for common beta- and gamma-emitting radionuclides for building-surface residual radioactivity (63 FR 64132, November 18, 1998).		
389	Chapter 7, Site Specific Decommissioning Costs	RG 1.179	Confirm the LTP includes the following: a. Estimate the decommissioning costs remaining at the time of LTP submittal. b. Compare the estimated remaining costs with the present funds set aside for decommissioning.		
390	Chapter 7, Site Specific Decommissioning Costs	RG 1.179	Confirm the decommissioning cost estimate evaluates the following seven cost elements, which are not meant to be all-inclusive: (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, (6) estimated final survey costs, and (7) estimated total costs.		

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP (page and paragraph)
391	Chapter 7, Site Specific Decommissioning Costs	RG 1.179	Confirm the cost estimate focuses on: • the remaining work and • provide details for each activity associated with the decommissioning, including the costs of labor, materials, equipment, energy, and services.		
392	Chapter 7, Site Specific Decommissioning Costs	RG 1.179	Confirm the cost estimates are based on credible engineering assumptions that are related to all remaining major decommissioning activities and tasks.		
393	Chapter 7, Site Specific Decommissioning Costs	RG 1.179	Confirm the cost estimate includes: • 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," issued January 2013, provides information on estimating waste disposal costs.		
394	Chapter 7, Site Specific Decommissioning Costs	RG 1.179	Confirm the cost estimate includes no credit for the salvage value of equipment.		

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP (page and paragraph)
395	Chapter 7, Site Specific Decommissioning Costs	SRP	Confirm the LTP decommissioning cost estimate includes an evaluation of the following cost elements: • cost assumptions used, including a contingency factor (normally 25 percent) • major decommissioning activities and tasks • unit cost factors • estimated costs of decontamination and removal of equipment and structures • estimated costs of waste disposal, including applicable disposal site surcharges and transportation costs • estimated final survey costs • estimated total costs		
396	Chapter 7, Site Specific Decommissioning Costs	SRP	Confirm the LTP focuses on detailed activity by activity cost estimates.		
397	Chapter 7, Site Specific Decommissioning Costs	SRP	Confirm the LTP also compares the funds available for decommissioning with the calculated total cost from the licensee's detailed cost analysis. In addition, Regulatory Guide 1.159, "Assuring the Availability of Funds for Decommissioning Nuclear Reactors," explains in detail the methods for estimating decommissioning costs, as well as accepted financial assurance mechanisms.		
398	Chapter 7, Site Specific Decommissioning Costs	SRP	Confirm the LTP cost estimate is based on credible engineering assumptions, and the assumptions are related to all major remaining decommissioning activities and tasks and are consistent with the information identified in Sections A.3 Identification of Remaining Site Dismantlement Activities and Remediation Plans and A.4 Final Radiation Survey Plan of this SRP.		

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP (page and paragraph)
399	Chapter 7, Site Specific Decommissioning Costs	SRP	Confirm the LTP cost estimate includes the cost of the remediation action being evaluated, the cost of transportation and disposal of the waste generated by the action, and other costs that are appropriate for the specific case. The current version of NUREG-1307, “Report on Waste Burial Charges,” provides guidance on estimating waste disposal costs		
400	Chapter 7, Site Specific Decommissioning Costs	SRP App A	Confirm the LTP cost estimate includes: • cost assumptions used, including a contingency factor and basis for each • cost estimate addressing the major decommissioning activities and tasks and their relationship to remaining dismantlement activities • description of the unit cost factors • estimated costs of decontamination and removal of equipment and structures • estimated costs of waste disposal, including applicable disposal site surcharges • estimated transportation costs • estimated final survey cost		
401	Chapter 8, Supplement to the Environmental Assessment (EA)	RG 1.179	Confirm the EA supplement describes in detail the environmental impact of the site-specific termination activity.		
402	Chapter 8, Supplement to the Environmental Assessment (EA)	RG 1.179	Confirm the EA supplement compares 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).		

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP (page and paragraph)
403	Chapter 8, Supplement to the Environmental Assessment (EA)	RG 1.179	Confirm the EA supplement analyzes the environmental impact of the site-specific activity. Include alternative actions and any mitigating actions.		
404	Chapter 8, Supplement to the Environmental Assessment (EA)	SRP	Confirm the EA supplement describes changes to the data that have arisen since the licensee submitted its “Applicant’s Environmental Report - Operating License Stage” or its “Applicant’s Environmental Report - Operating License Renewal Stage,” as appropriate.		
405	Chapter 8, Supplement to the Environmental Assessment (EA)	SRP	Confirm the EA supplement describes the potential environmental impacts associated with site specific termination activities from the time the LTP is submitted until the license is terminated.		
406	Chapter 8, Supplement to the Environmental Assessment (EA)	SRP	Confirm the EA supplement states the licensee’s determination regarding whether the activities and effects are bounded by the potential impacts described by any site-specific EIS or EA developed in support of licensing the facility, NUREG-0586 as supplemented, or the PSDAR. The EA supplement should also describe any proposed mitigation measures the licensee will take to avoid significant impact.		
407	Chapter 8, Supplement to the Environmental Assessment (EA)	SRP	Confirm the EA supplement identifies the parts, if any, of the facility or site that were released for use before approval of the license termination plan.		
408	Chapter 8, Supplement to the Environmental Assessment (EA)	SRP App A	Confirm the EA supplement describes any new information or potential significant environmental impact(s) associated with the site-specific termination activities related to the end use of the site (the environmental evaluation does not have to address decommissioning activities but focuses on site end use)		

ACRM Seq #	Chapter	Source	Description	Addressed in LTP (Yes/No)	Location in LTP (page and paragraph)
409	Chapter 9, Portions of Facility Released Prior to LTP Approval	RG 1.179 1.h	Confirm identification of parts, if any, of the facility that were released for use before approval of the LTP under 10 CFR 50.82(a)(9)(ii)(H).		

APPENDIX D. SUGGESTED FEDERAL AND STATE REGULATORY INTERFACE PLAN

D.1 Introduction

The NRC staff encourages applicants to meet with the staff to discuss potential license actions. In this way, the staff and applicants gain a mutual understanding of the purpose and type of future actions. This recommendation is especially true for complex licensing actions such as a License Termination Plan, the implementation of the plan and the final status survey process.

This appendix provides a suggested plan to interface with the NRC and State regulators to ensure each party is fully engaged every step of the way from initial preparations of the LTP through the end of the decommissioning process resulting in either Part 50 license termination or in a partial site release to reduce the Part 50 license to the ISFSI area.

The plan provides timelines for each section below to provide a goal of completing the entire process in a realistic and timely manner. Each section below is further described in this plan:

- LTP Development and Review
- LTP Changes
- FSS Development and Review
- FSS Report Changes
- Compliance Discussion and Site Release

D.2 LTP Development and Review

Early and frequent discussions between U.S. Nuclear Regulatory Commission (NRC) staff and licensees are encouraged during the planning and scoping phase supporting the preparation of the LTPs. In this context, a licensee should schedule a series of meetings with the NRC Project Manager assigned to the site to discuss the planning and content of the LTP. It is recommended separate meetings are held in the development process to discuss the topics as grouped below:

- Chapters 1, Introduction; 3, Remaining Dismantlement Activities; 7, Site Specific Decommissioning Costs; and 8, Supplement to the Environmental Report
- Chapter 2, Site Characterization
- Chapter 4, Remediation Plan including NRC Site Involvement
- Chapter 5, Final Radiation Survey Plan and Report Content
- Chapter 6, Compliance with Radiological Criteria for License Termination and Confirmatory Surveys including State Involvement

Pre-application public meetings are a valuable component of work planning and minimize unnecessary effort and rewrites. Staff's experience has shown that these types of meetings reduce the number of requests for additional information (RAI) and provide a good background for reviewers. Information that is typically discussed during these meetings includes the type of and need for action, proposed schedules, licensing basis, and methodology. No regulatory decisions are made during these meetings, but this opportunity allows for an exchange of information and can aid in streamlining the review.

Typical LTP development is likely to take 18 to 24 months. During this time the historical site assessment and initial site characterization is completed. Information collected includes among other topics: past and current licensed operations; types and quantities of radioactive materials used or stored; activities (current or past) that may have an impact on decommissioning operations and initial site surveys and samples. This information is part of the discussion for LTP Chapter 2 and is recommended to be discussed early with the NRC in the LTP development process.

The next important pre-application meeting should focus on the remediation plan and NRC involvement during remediation. The NRC regional inspectors are typically on site during the remediation process and follow decommissioning progress closely. Licensees should also encourage NRC headquarters staff involvement from the Division of Decommissioning, Uranium Recovery and Waste Programs. The NRC reviewers assigned to review the LTP should visit the site when remediation surveys or final status surveys are performed for remaining subsurface structures prior to those structures being back-filled, and will minimize questions raised by the staff as to the adequacy of remediation or adequacy of the surveys performed long after the field work is completed and the final status survey report is under their review. Obtaining NRC agreement on their level of involvement is critical to reducing RAIs and maintaining regulatory confidence during decommissioning.

The next important meeting is on the format and content of the final status survey reports. The recommended format and content provided in this document should be followed as it is the recommended approach for the nuclear power industry. The licensee should challenge requests to see field data or other in-process data in general, and only provide such data by exception for any survey result location(s) that are questionable and may require further analysis or resurvey.

Another pre-application meeting is recommended for the compliance calculation, end-state conditions and partial site release or license termination. It is important for all parties to agree on the end state conditions and how any more conservative end state conditions required by the state or property owner will be met. The final licensing actions to terminate or reduce the Part 50 licensed footprint need to be discussed.

Finally, an additional meeting should be held to discuss license termination costs and environmental impacts. Particular attention should be paid to site specific impacts to be evaluated. Due to the age of many plants, state historical impacts need to be fully considered working closely with the state historic preservation office. This is particularly important early on in the LTP process if there are historic or archaeological sites or threatened or endangered species on the property that need to be protected during decommissioning. Mitigation measures need to be agreed to if a site or species will be adversely impacted.

Before meeting with the NRC staff, a licensee is encouraged to prepare an agenda that identifies technical elements that are applicable (based on a preliminary review); areas that require clarifications from the NRC staff before decisions can be made as to their applicability to the site or facility; and scope and level of technical details addressing technical elements and regulatory requirements.

During the meeting, the NRC staff and licensee representative would go over each item of the agenda and address specific questions. NRC would present an overview of its review process, including discussions of the project schedule and major milestones. The product of the meeting is a set of meeting minutes that define the technical elements and regulatory requirements to be covered in the LTP submittal. This process will result in a better understanding of the type of information to be included in the document and to familiarize the licensee with the process that the staff will use to evaluate the information contained in the LTP. This approach is expected to minimize the need for requests for additional information, reduce the number of iterations and submittals, and expedite the staff's technical review.

D.3 Submittal and Public Meeting

Once the LTP is submitted, the NRC will begin the acceptance review process lasting up to 90 days.

The NRC regulations currently offer the public several opportunities to review and provide comments on licensee documents during the decommissioning process. Specifically, under the NRC regulations in Title 10 of the Code of Federal Regulations (10 CFR) Part 50.82, the NRC is required to publish a notice of the receipt of the licensee's License Termination Plan (LTP), make the LTP available for public comment, schedule separate meetings in the vicinity of the location of the licensed facility to discuss the LTP within 60 days of receipt, and publish a notice of the meetings in the Federal Register and another forum readily accessible to individuals in the vicinity of the site

The NRC public meeting will discuss the LTP review and approval process including review of the final status survey reports and the request for license termination and partial site release. The NRC will also provide notice in the Federal Register for public comment and any requests for a hearing. It is important to have resolved any comments or concerns from state regulators during the LTP development and pre-application meetings to ensure the state does not feel it necessary to request a hearing to ensure compliance with state regulations.

D.4 NRC Review of LTP

This process has historically taken from 2 to 4 years with multiple rounds of RAIs. The mutual goal of the industry is to complete NRC review and approval of the LTP in 2 years and complete NRC review of Final Status Survey Reports in 4 to 6 months after licensee submittal of each package of FSS Reports.

To achieve this goal, it is imperative on the licensee to obtain an NRC schedule for LTP review and to conduct periodic (quarterly) meetings with the NRC staff to ensure questions and issues are being resolved as the review progresses. Obtaining agreement to submit supplemental information and LTP changes on the docket during the review process will aid in the review process.

Requests for Additional Information (RAI) should be expected for this complex document during the review process. It is recommended to establish a document sharing process for large documents that are blocked by email firewalls. The RAI process should follow the company established process for RAIs for other license amendment or NRC approval requests.

D.5 EPA and NRC Memorandum of Understanding

The Environmental Protection Agency (EPA) and the NRC, in recognition of their mutual commitment to protect the public health and safety and the environment, entered a Memorandum of Understanding

(MOU) on October 9, 2002 (Reference D-1), in order to establish a basic framework for the relationship of the agencies in the radiological decommissioning and decontamination of NRC-licensed sites.

The MOU is intended to address issues related to the EPA involvement under CERCLA in the cleanup of radiologically contaminated sites under the jurisdiction of the NRC. EPA will continue its CERCLA policy of September 8, 1983, which explains how EPA implements deferral decisions regarding listing on the National Priority List of any sites that are subject to NRC's licensing authority. The NRC's review of sites under NRC jurisdiction indicates that few of these sites have radioactive ground-water contamination in excess of the EPA's MCLs. At those sites at which NRC determines during the license termination process that there is radioactive ground water contamination above the relevant EPA MCLs, NRC will consult with EPA and, if necessary, discuss with EPA the use of flexibility under EPA's phased approach to addressing ground water contamination. NRC has agreed in the MOU to consult with EPA on the appropriate approach in responding to the circumstances at sites where ground-water contamination will exceed EPA's MCLs, NRC contemplates either restricted release or the use of alternate criteria for license termination, or radioactive contamination at the time of license termination exceeds the MCL concentrations.

In addition to NRC actions to be taken if a site has groundwater concentrations that exceed the EPA's MCLs, the MOU lists trigger levels for soil contamination that would cause NRC to consult with EPA for planned or actual residual radioactivity above the trigger levels at a site. These trigger levels are based on either residential and industrial/commercial soil concentrations and are provided in Table 1 of NUREG-1757, Volume 1, Revision 2 (Reference 1-4) along with additional details on the EPA/NRC MOU. It should be noted that some of these trigger levels are below the NRC screening values or may be below site specific DCGLs determined for an individual site. Licensees should consider specific options when the consultation trigger level(s) are lower than the soil DCGLs to be used at a site including performing:

- Performing the FSS using the trigger levels as Operational Soil DCGLs when they are lower than those determined by dose modeling and/or after adjustment for other media such as dose due to groundwater

- Performing additional remediation when actual soil concentrations exceed the trigger levels but not the Operational Soil DCGLs determined by dose modeling and/or after adjustment for other media

D.6 LTP Implementation and Changes

It is common during the decommissioning process that an LTP change is required. Many changes can be implemented without NRC review and approval based on the criteria in the LTP license condition. Any changes requiring NRC review and approval are submitted as license amendment requests and the NRC usually agrees to an expedited review schedule (90 days or better) so as not to slow down decommissioning progress.

It is important to note that the LTP is incorporated by reference into the FSAR or its equivalent, a Decommissioning Safety Analysis Report (DSAR). It is recommended the LTP updates be submitted on the same schedule as the DSAR updates required under 10 CFR 50.71.

D.7 Final Status Surveys

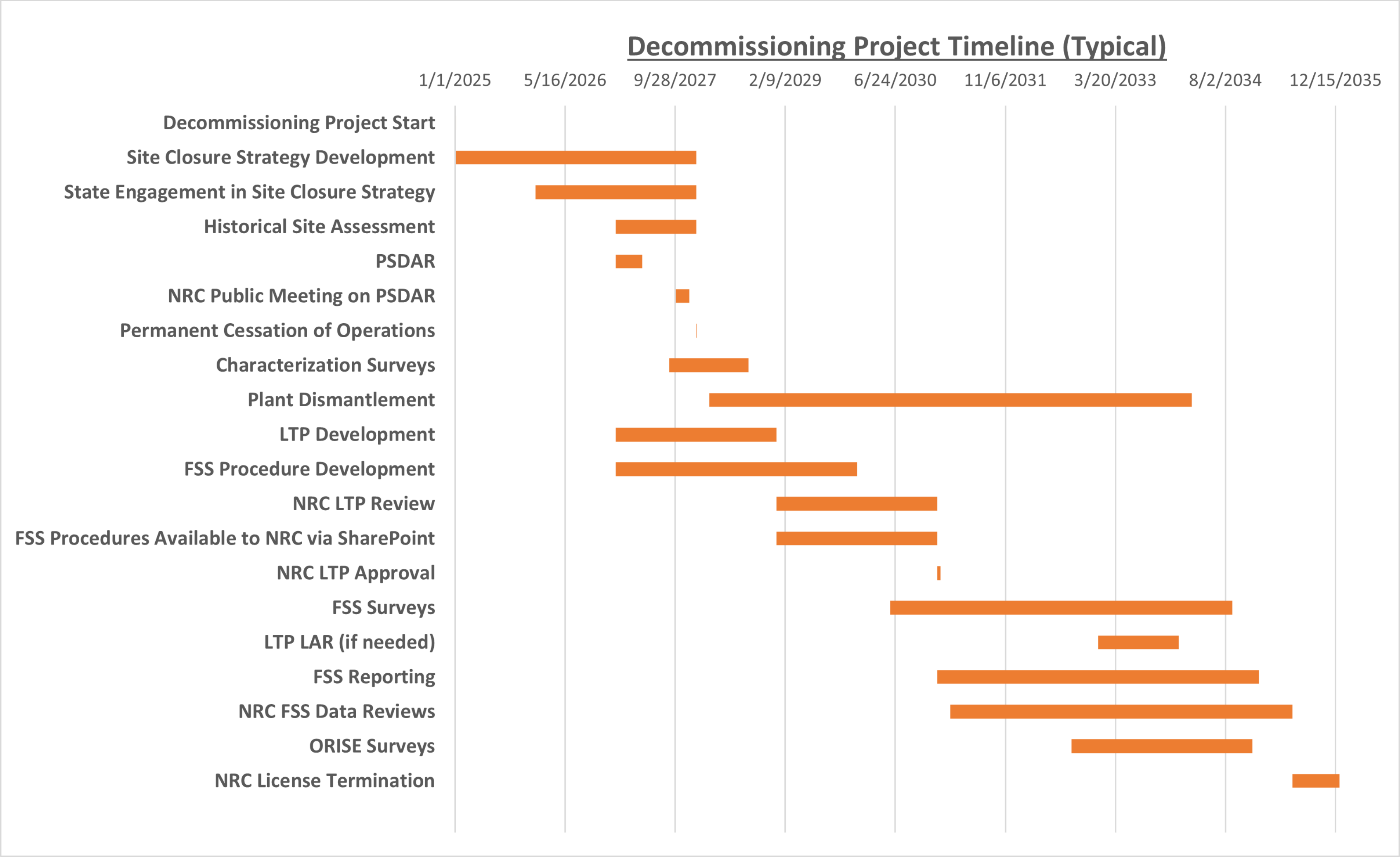
During decommissioning, final status surveys are performed on areas that have completed decommissioning activities. Very often, these surveys are performed before the LTP has been approved for some areas such as the Turbine Building or outer buildings (offices, warehouses, etc.). Performing these surveys at risk should be well understood in particular with the selection of the DCGLs to be used in developing these final status surveys. For example, some survey areas may be subject to operational DCGLs as described in Chapter 5 and Appendix B and the selection of these DCGLs may require site parameters and end-states to be well defined and may not be known at the time of these final status surveys. Additionally, the selection of ROCs is also an important element of the LTP which will require NRC acceptance. Therefore, conservative decisions of the likely ROCs should be made prior to LTP approval.

It is recommended that NRC review of the survey plan be formally documented within 30 days of submittal to provide confidence that the plan will not be questioned as to its validity during the FSS Report review. If the NRC Project Manager cannot meet the 30-day document review schedule, then NRC review of the plan by the Regional Inspector, including NRC site inspection of the survey plan implementation, should be documented in the applicable inspection report. Licensees should request the NRC to explain how the NRC's regional inspections and confirmatory surveys will be utilized during their acceptance reviews.

D.8 References for Appendix D

D-1 Memorandum of Understanding between the Environmental Protection Agency and the Nuclear Regulatory Commission, Consultation and Finality on Decommissioning and Decontamination of Contaminated Sites, dated October 9, 2002.

APPENDIX E. TYPICAL LICENSE TERMINATION MILESTONE SCHEDULE



APPENDIX F. SITE SPECIFIC DOSE MODELING EXPERIENCES

This appendix provides the reader with case studies concerning the development of site-specific building, soil, and groundwater Derived Concentration Guidance Levels (DCGLs) for sites that have been decommissioned. Chapter 6 provides an overview of the development process and information on NRC generic screening DCGLs. A site may wish to develop site-specific DCGLs for reasons such as the following:

- Certain characteristics of the site make the NRC screening values very conservative for that site (i.e., very deep groundwater table beneath the site, soil with relatively high K_ds that make radionuclides present in the soil relatively immobile.)
- Long term future use of the site is known or expected to be industrial, which will likely make certain dose pathways included in the development of the NRC screening values not applicable.

As discussed in Chapter 6, the NRC recommended the use of realistic exposure scenarios that reflected expected use of the site for the foreseeable future. The dose pathways to be analyzed are then set based on a realistic future use of the site. In order to illustrate actual plant experience before and after the updates to the NRC guidance, actual cases for power plant sites being decommissioned are presented below. The case involving Connecticut Yankee (CY) will show the experience for a complex site performing dose modeling calculations prior to the update to NRC guidance issued in 2003-2004. The Rancho Seco plant experience shows the use of the realistic scenario approach discussed in the updated NRC guidance. Case studies from the Big Rock Point, Maine Yankee and Trojan plants in the late 1990s and early 2000s are also included in order to provide alternate approaches. Additionally, to show the impact of more recent NRC guidance, experiences from the Zion, LaCrosse and Fort Calhoun plant sites are also summarized.

F.1 Connecticut Yankee Experience

Due to the presence of elevated groundwater radionuclide contamination at Connecticut Yankee (CY), dose from contamination in as many as three different media needed to be accounted for in certain land areas. These media were soil, groundwater and subsurface concrete. Most of the information in this section concerning CY was obtained from The CY License Termination Plan (Reference F-1).

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

Dose Total = Dose Soil + Dose Existing Groundwater + Dose Future Groundwater **Equation F-1**

Definitions of the terms of this equation are as follows:

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

- Dose Soil: The portion of the dose from all pathways that is contributed by the soil related pathways.

- Dose Existing Groundwater: The portion of the dose from all pathways from residual radioactivity currently in groundwater on site that would still be present at the time of release of the site.

- Dose Future Groundwater: The portion of the dose from all pathways due to residual radioactivity that is projected to leach from the concrete buildings at CY and be present in groundwater at the time of site release. A discussion of the development of this dose calculation methodology is provided later in this appendix. It should be noted that the maximum dose due to residual radioactivity leaching from concrete buildings doesn't have to occur at the time of site release. Rather, residual radioactivity can leach to groundwater within the 1000-year compliance period. The projected dose from this leaching as various times after license termination should be evaluated and compared to the site release criteria along with the dose from other media.

It should be noted that some of the components of Equation F-1 may not apply at all power plant sites or all locations on a site. For example, the "Existing Groundwater" dose component may not apply in scenarios where the use of the site was controlled to preclude the use of groundwater or the groundwater was of poor quality that precludes consumption. The "Future Groundwater" component may not apply if no contaminated concrete in contact with groundwater was left on site at license termination. The "Future Groundwater" component may also not apply if groundwater sampling has been conducted for a sufficient period of time after concrete remediation to detect the leaching of radionuclides from the concrete into the groundwater. In such a case, the detected groundwater contamination from the concrete would be included in the "Existing Groundwater" dose component. In order to meet this compliance equation to satisfy the NRC and the Connecticut site release criteria, the following methodology was used at CY:

- DCGLs corresponding to 25 mrem/yr were determined for each of the media as discussed later in this report.

- Target levels were set for each type of media by dividing up the total dose for the site release criteria of 19 mrem/yr TEDE [Note: Connecticut required this more stringent site criterion, versus the NRC criteria of 25 mrem/yr. For sites using the NRC criteria, the same approach could be used but setting the targets based on a total dose of 25 mrem/yr. It should also be noted that, as was the case at Connecticut Yankee, any site release criteria agreed to with state regulators or other stakeholders, should not be included in the LTP submitted to the NRC for review and approval. To include such criteria in the LTP would require the NRC to enforce those criteria. As the final status survey for the land areas that were impacted by "existing" and "future" groundwater dose needed to be conducted beginning in May 2006, these target levels were set based on the projections of what the actual doses would be at the time of site release (expected to be in mid-2007). Sufficient post remediation (the important remediation had taken place from 2004 to 2005) groundwater and concrete sample data had been collected to allow the confident setting of the dose targets for the various media as follows:

    - The "existing groundwater" dose target was set at 2 mrem/yr TEDE using the Groundwater DCGLs for the areas affected by groundwater contamination. This dose was set based on groundwater monitoring sample results and to bound the groundwater concentrations expected to exist at the time of release of the site (projected to be mid- 2007).

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

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

The following sections provide summary descriptions on various dose scenarios used to calculate release limits for the various media at CY. Sections are provided for each media type and include experiences for sites other than CY to illustrate the evolution of dose modeling over time and show the impact of revised NRC guidance.

F.2 Dose Assessment Modeling for Land Areas – Late 1990s to Early 2000s

F.2.1 Connecticut Yankee

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.

It should be noted that more recent NRC guidance has been provided in NUREG-1757, Volume 2, Revision 2 and DWUP-ISG-02. A summary of this guidance concerning parameter selection is provided in Section 6.1.2.

F.2.2 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. Section 6.1.6 discusses the assumptions used in determining land area DCGLs for the Big Rock Point Modified Resident Farmer Scenario. 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.2.3 Rancho Seco Experience

As discussed in Section 6.1.6, 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 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 the impact of existing groundwater concentrations in adjusting soil DCGLs or to determine separate Groundwater DCGLs as was done at Connecticut Yankee (discussed below). Rancho Seco used RESRAD Version 6.3 (released in 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.

Rancho Seco 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 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-6) 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-7) as follows:

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

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

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

Calculation of Surface Soil DCGLs for Rancho Seco

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.

Applicability of Surface Soil DCGLs to Sub-Surface Soil for Rancho Seco

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.

Ranco Seco 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 the contaminated soil material;
- Internal dose from inhalation of airborne radionuclides; and
- Internal dose from ingestion of:
    - Drinking water from a contaminated well, and
    - Contaminated soil.

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

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

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

Calculated Dose for a Resident Farmer Scenario for Rancho Seco

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 would 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 transfer of all or any portion of the impacted area of the site to another entity immediately upon the release of all but the Interim Spent Fuel Storage Installation (ISFSI) from the NRC license. At the time the DCGL analysis was performed, this partial release of the site was targeted to occur on July 1, 2008. Thirty years following the release of non-ISFSI areas of Rancho Seco from the NRC license was considered to be a reasonable time period during which the District would not transfer all or any portion of the impacted area of the site to another entity. This was reasonable since SMUD had made capital investment in new construction on the site. If the site was to be transferred to another entity 30 years after the release (i.e., July 1, 2038) and a subsistence farm was established on the site, the dose to this Resident Farmer would not exceed 25 mrem/yr. 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 at a later date). It shows how a combination of radioactive decay and the redistribution of contamination due to precipitation and groundwater flow can reduce the dose effect from a certain concentration of radionuclides in soil. It should be noted that 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.

F.3 Dose Modeling for land Areas - Decommissioning Projects After 2010

F.3.1 Zion Resident Farmer Scenario (Reference 5-2)

For the Zion site the soil DCGLs were calculated using the standard RESRAD parameter selection and uncertainty analysis methods as defined in NUREG 1757. 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 was 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.

Zion Alternate Scenarios

For the Zion site, several alternate scenarios for land use after backfill were qualitatively considered including industrial use, recreational use (i.e., parkland), and residential use without a water supply well or onsite garden. The Resident Farmer scenario with onsite well is clearly a very conservative, bounding scenario relative to the alternatives.

The SSDM (discussed in later in this Appendix) and the alternate scenarios considered above are based on the “as left” geometry of the residual contamination in the backfilled Basements. Two additional low probability alternate scenarios were considered that included changes to the “as left” backfilled geometry. The first entails construction of a basement to the Resident Farmer house within the fill material. Zion noted that the assumed three meter depth of the basement excavation is insufficient to encounter fill material potentially containing residual radioactivity (resulting from leaching of residual radioactivity from surfaces after backfill) assuming the basement is not constructed within the saturated zone. However, a simple check of direct radiation dose to the resident was conducted to confirm the expectation that the dose would be negligible.

The second alternate scenario that includes disturbance of the as-left geometry considers a very unlikely assumption of a large-scale excavation of the backfilled structures after license termination. The potential doses from large scale excavation were checked by averaging the hypothetical maximum total activity corresponding to 25 mrem/yr in the SSDM over the mass of the basement concrete and fill. The average concentrations were compared to the soil concentrations equivalent to 25 mrem/yr based on an industrial use scenario which was assumed to be the only future use that would justify large scale excavation of fill and concrete located deep within the saturated zone.

A third alternate scenario was evaluated which assumed that the drill for a water well encounters penetrations, embedded pipe, and basement surfaces assuming that no activity is released to the fill. The activity in the penetration, embedded pipe or basement surface that is captured by the drill is assumed to be brought to the surface in the drilling spoils. Two receptors were evaluated; the resident farmer and a worker.

The alternate drilling spoils scenario dose for all embedded pipe, penetrations and basement surfaces except for the Steam Tunnel Floor Drains was calculated to be below 25 mrem/yr. The DCGLs for the Steam Tunnel Floor Drains were reduced by a factor of 2.89 (71.16/25) which reduces the maximum dose to 25 mrem/yr.

F.3.2 LaCrosse Land Area Scenarios (Reference 6-6)

As the LaCrosse site contains 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 F-2) 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 F-10) 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 m^3/hr. This was applied to an on-site time of 2190 hrs/yr to yield a parameter value of 3066 m^3/yr.

LaCrosse Alternate Scenarios

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

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 that assumed for the industrial worker.

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 was 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. For conservatism, LaCrosse adjusted the soil and basement DCGLs used for FSS to ensure that no alternate scenario dose could exceed 25 mrem/yr.

F.3.3 Fort Calhoun

As summarized in Section 6.2.2, a review of area land use identified six reasonably foreseeable scenarios. The land use scenarios, environmental pathways, exposure pathways and associated average member of the critical group (AMCG) are summarized in Table 6-4 and Table 6-5. The aquatic pathway from an onsite pond was considered not credible due to engineering and cost issues of construction and proximity to the Missouri River which negates any foreseeable need. Therefore, ingestion of aquatic foods is not included as an exposure pathway in any of the land use scenarios. The following summarizes the information on dose modeling contained in the Fort Calhoun LTP.

Fort Calhoun Compliance Exposure Scenario

A qualitative analysis of the reasonably foreseeable scenarios in Table 6-4 concluded that the residential farming scenario would result in the highest dose. All other scenarios either have environmental pathways removed or lower occupancy times and food ingestion rates. The one possible change to the impact of an environmental pathway in the various scenarios is the well water concentration when the well pumping rate is lower or higher. For example, the residential scenario would have a lower well pumping rate than the residential farming scenario due to decreased irrigation. The recreational and industrial scenarios could have higher pumping rates. The pumping rates were evaluated through sensitivity analysis and the well water concentrations were found to be insensitive to pumping rate.

The bounding residential farming exposure scenario was applied for Fort Calhoun to demonstrate compliance with the dose criterion for soil, backfilled basements (including embedded pipe and penetrations), backfill soil, buried pipe, and existing groundwater. The bounding building occupancy exposure scenario is applied to demonstrate compliance for above ground buildings. The doses attributable to each media - soil, backfilled basements (including embedded pipe and penetrations), backfill soil, buried pipe, above ground buildings, and existing groundwater - are summed to ensure that the compliance dose is prudently conservative.

The evaluation and results for these various scenarios are summarized later in this appendix.

Resident Farmer Dose Modeling for Fort Calhoun

The AMCG is a resident farmer that resides in a house constructed on the site and spends the majority of the year onsite conducting subsistence farming activities. The contamination is transported from soil to plants and from plants to animals. The contaminated plants, meat and milk are subsequently ingested by the AMCG. Contamination is also transported and ingested through hand to mouth behavior by the AMCG. Groundwater that becomes contaminated through the infiltration of precipitation and irrigation is captured by an onsite well and used for drinking water and irrigation (see environmental and exposure pathways in Table 6-5).

Fort Calhoun Station (FCS) is situated on low flat land bordering the Missouri River. Because of its location on the flood plain of the river, the topography in the area of the power plant is relatively flat. The resulting hydraulic gradient within the unconsolidated sediments in the flood plain is also relatively flat. This low hydraulic gradient, combined with moderate hydraulic conductivity of the generally fine-grained alluvial aquifer material, results in relatively slow ground water flow velocity beneath the site. The site-specific hydraulic gradient in $8.4x10^{-4}$. The unsaturated zone soil type is silt loam. The saturated zone soil type is sand. The site-specific hydraulic conductivities are 34.4 m/yr and 4350 m/yr, for unsaturated and saturated zones, respectively.

Soil analyses were conducted to determine site-specific values for density, total porosity, effective porosity, and field capacity. The values for the soil parameters are provided in Attachment 6.1 of the Fort Calhoun LTP which also contains values for the remaining physical parameters not mentioned here, such as irrigation and precipitation, and the metabolic and behavioral parameters of the AMCG. The size of the conceptual site is assumed to be the total area of the FCS Class 1 open land areas, i.e., 79,600 m^2.

Onsite water levels collected from 17 wells in June 2020 ranged from 10.5 to 8.5 feet below ground surface (bgs) (3.1 to 4.6 m). The HSA states that the water table ranges from 2 to 20 ft bgs. However, the 2-foot bgs value is expected to correspond to a high stage of the Missouri River and therefore to occur infrequently. The CSM states that a water table elevation range of 10 to 15 ft bgs is reasonable based on the review of site-specific measurements and Missouri River stage data. The conceptual site model conservatively assumes that the water table elevation is 1.1 m bgs as discussed below.

The vadose zone thickness at the site is assumed to be 1.1 m which was selected initially to accommodate a conceptual soil source term thickness of 1 m with a nominal unsaturated zone thickness of 0.1 m. Based on the groundwater elevations at the site as discussed above, it is not unreasonable to assume that the groundwater table will periodically be as high as 1.1 m bgs although the average levels are expected to be in the range of 10-15 ft bgs. The 1.1 m vadose zone thickness is also applied in the conceptual models for the other contaminated media (basement concrete [and Containment liner], embedded pipe, and buried pipe). Assuming a water table elevation at the high end of the site range (1.1 m bgs) is conservative because it reduces transport time and minimizes the effect of dispersion and decay in the calculation of groundwater concentrations. In addition, a water table at 1.1 m bgs results in the source terms for the in situ buried pipe scenario, in situ basement concrete scenario, and embedded pipe scenario being fully submerged in groundwater which increases the groundwater concentration compared to a source term that is contained in the vadose zone.

The RESRAD ONSITE Version 7.2 (RESRAD) code contains conceptual models and corresponding mathematical formulations that are compatible with the FCS conceptual model and includes dose calculations for all of the exposure pathways listed in Table 6-5 for land areas. RESRAD was utilized to calculate soil DCGLs for Fort Calhoun in much the same manner as was done for other plants discussed above. Some of the site-specific information for Fort Calhoun used in the model is discussed below.

The soil contamination is assumed to be uniformly distributed at depths of 1 m and 0.15 m. The 1 m thickness is included along with the standard 0.15 m to address the possibility that soil contamination is identified during FSS that is between 0.15 and 1 m thick. In both cases, the unsaturated zone depth is 0.1 m. The contaminated area size is conservatively assumed to be 79,600 m^2 which includes the entire FCS Deconstruction Area (DA) and the waste haul path and loading area (the FCS Class 1 areas). The waste haul path area is assumed to be contiguous with the DA for the purpose of selecting the contaminated area size.

There are two source term release pathways: resuspension to air and leaching to groundwater. Release of radionuclides by water is assumed to be a function of a constant infiltration rate, constant moisture content, and equilibrium desorption. The release of radioactivity to the unsaturated zone and groundwater is controlled by infiltration, driven by precipitation and irrigation, through the unsaturated soil from the ground surface down to the water table and by groundwater flow at depths below the water table. The partitioning of contaminants between the solid and aqueous phase is assumed to be a linear process controlled by the distribution coefficient. The airborne release is accounted for by assuming a constant mass-loading of respirable particles in air with resuspension and deposition in equilibrium.

Fort Calhoun Soil RESRAD Input Parameters for Uncertainty Analysis

Uncertainty analysis was performed to ensure that conservative values are selected for parameters that have a relatively high correlation to dose. Table F-1 provides the input parameters used to perform the uncertainty analysis for the soil dose assessment. The deterministic parameter selection process used was the same manner as is described for Rancho Seco in Section 6 except that a few parameters were assigned site-specific probability density functions (PDFs). Additionally, the Partial Rank Correlation Coefficient (PRCC) cutoff value (absolute value of 0.25) for categorizing parameters was not universally applied. Each run is judged by the analyst. There are cases where the PRCC results are all below, but near, the |0.25| PRCC threshold, with consistent results for all three of the repetitions. In these cases, at the discretion of the analyst, a single parameter sensitivity analysis is used to augment the PRCC results. Using these additional methods, some parameters that do not exceed the |0.25| threshold are also designated as sensitive. Parameters that are designated as sensitive when the PRCC is below |0.25| are identified in Table F-1, Table F-2, and Table F-3.

For distribution coefficients (K_d), the soil-specific PDFs listed in Reference F-15, Tables 2.13.1, 2.13.2, and 2.13.10 are used for the uncertainty analysis based on the site-specific soil types. The contaminated zone and unsaturated zone are both silt loam soil type. The soil type for the saturated zone is sand. The median of the K_d distribution is applied as the deterministic value for the uncertainty analysis but the final K_ds applied in the DCGL calculation are the deterministic values selected through uncertainty analysis.

For simplicity and to ensure that the calculated dose is conservative for all radionuclides, all plant/meat/milk transfer factors were assigned the 75th percentile of the NUREG-6697 (RESRAD default) distribution.

The uncertainty analysis was performed for each radionuclide individually. This conservatively disregards the reduced influence of low abundance radionuclides on the total dose and eliminates the potential impact of uncertainty in radionuclide mixture fractions. In addition, the uncertainty analysis is performed for a 1 m thickness of primary contamination. The results are also applicable to the 0.15 m contamination thickness.

Fort Calhoun Soil Uncertainty Analysis Results and Deterministic Parameter Selection

The results of the soil uncertainty analysis and the selected deterministic parameters are provided in **Error! Reference source not found.**, Table F-2, and Table F-3. When a parameter in Table F-1 is identified as sensitive for one or more radionuclide, the selected deterministic value is assigned to all radionuclides, including those that are not sensitive to the given parameter. The majority of sensitive parameters have a consistent correlation with several radionuclides. The most obvious example is root depth which is negatively correlated due to the thickness of the contaminated zone. Another example is parameters related to inhalation that are consistently sensitive for certain actinides. The contaminated zone and unsaturated zone soil types, and corresponding K_ds, are the same. If either is sensitive, then both are changed for consistency. There were no conflicting results in contaminated zone and unsaturated K_d correlations, i.e., one positive and one negative.

Additional information on the process used to select the input parameters to RESRAD is included in the Fort Calhoun LTP.

Table F-1: Soil RESRAD Parameter Uncertainty Analysis Results for Non-Nuclide Specific Parameters that were Sensitive and Selected Deterministic Values

Parameter	Correlation to Dose	Radionuclide	Basis of Deterministic Parameter Selection	Selected Deterministic Value
Contaminated zone erosion rate (m/yr)	negative	Pu-241	25^{th} percentile	7.59E-04
Contaminated zone b parameter (unitless)	negative	H-3	25^{th} percentile	2.87
Evapotranspiration coefficient (unitless)	positive	C-14, Eu-152, Sb-125, Tc-99	75^{th} percentile	0.87
Wind Speed (m/s)	negative	Am-241[1], C-14, Cm-243	25^{th} percentile	3.27
Runoff coefficient (unitless)	positive	H-3, Tc-99	75^{th} percentile	0.63
Depth of roots (m)	negative	Am-241, C-14[1], Ce-144, Cm-243, Co-58, Co-60, Cs-134, Cs-137, Eu-152, Eu-154, Eu-155, Fe-55, H-3, Ni-59, Ni-63, Np-237, Pu-238, Pu-239, Pu-240, Pu-241, Sb-125, Sr-90, Tc-99	25^{th} percentile	1.23
Well pump intake depth (m)	negative	H-3, Np-237, Pu-238, Tc-99	25^{th} percentile	21.4
Mass loading for inhalation (g/m^3)	positive	Am-241, Pu-238, Pu-240[1], Cm-243	75^{th} percentile	2.87E-05

Parameter	Correlation to Dose	Radionuclide	Basis of Deterministic Parameter Selection	Selected Deterministic Value
Indoor dust filtration factor (unitless)	positive	Am-241, Pu-238, Cm-243	75th percentile	0.75

1. PRCC is less than |0.25| but correlation is indicated. Confirmed to have slight correlation to dose by single parameter sensitivity analysis.

Table F-2: Soil RESRAD Parameter Uncertainty Analysis Results for K_d of Contaminated Zone and Unsaturated Zone and Selected Deterministic Values

Nuclide	Correlation to Dose	Basis of Deterministic Parameter Selection	Selected Deterministic Value (cm^3/g)
Am-241	NS[1]	50th percentile	1.25E+03
C-14	positive	75th percentile	96.7
Ce-144	NS	50th percentile	3.01E+03
Cm-243	NS	50th percentile	1.9E+04
Cm-244	NS	50th percentile	1.9E+04
Co-58	positive	75th percentile	5.05E+03
Co-60	positive	75th percentile	5.05E+03
Cs-134	NS	50th percentile	3.5E+03
Cs-137	NS	50th percentile	3.5E+03
Eu-152	positive	75th percentile	7.27E+03
Eu-154	positive	75th percentile	7.27E+03
Eu-155	positive	75th percentile	7.27E+03
Fe-55	NS	50th percentile	889
H-3	negative	25th percentile	4.3E-02
Ni-59	NS	50th percentile	179
Ni-63	positive	75th percentile	532
Np-237	negative	25th percentile	9.05
Pu-238	NS	50th percentile	953
Pu-239	NS	50th percentile	953
Pu-240	NS	50th percentile	953
Pu-241	NS	50th percentile	953
Sb-125	positive	75th percentile	128
Sr-90	positive	75th percentile	168
Tc-99	positive	75th percentile	1.47E-01

1. not sensitive (NS)

Table F-3: Soil RESRAD Parameter Uncertainty Analysis Results for K_d of Saturated Zone and Selected Deterministic Values

Nuclide	Correlation to Dose	Basis of Deterministic Parameter Selection	Selected Deterministic Value (cm^3/g)
Am-241	NS[1]	50th percentile	1000
C-14	NS	50th percentile	11
Ce-144	NS	50th percentile	399
Cm-243	NS	50th percentile	3,390
Cm-244	NS	50th percentile	3.390
Co-58	NS	50th percentile	260
Co-60	NS	50th percentile	260
Cs-134	NS	50th percentile	528
Cs-137	NS	50th percentile	528
Eu-152	NS	50th percentile	829
Eu-154	NS	50th percentile	829
Eu-155	NS	50th percentile	829
Fe-55	NS	50th percentile	321
H-3	NS	50th percentile	6.02E-02
Ni-59	NS	50th percentile	130
Ni-63	NS	50th percentile	130
Np-237	negative	25th percentile	5.49
Pu-238	NS	50th percentile	399
Pu-239	NS	50th percentile	399
Pu-240	NS	50th percentile	399
Pu-241	NS	50th percentile	399
Sb-125	NS	50th percentile	16.9
Sr-90	NS	50th percentile	22
Tc-99	NS	50th percentile	0.40

1. not sensitive (NS)

RESRAD was run using the deterministic parameter values given in the tables above as the input values to calculate the soil DCGLs. Table F-4 provides DCGLs with no adjustment for the dose from insignificant contributor (IC) radionuclides. The final soil DCGLs for the Radionuclides of Concern (ROC), after correction for IC dose, are provided in Table F-5.

Table F-4: Soil Initial Suite DCGLs (No IC Dose Adjustment)

Radionuclide	Soil DCGL 0.15 m (pCi/g)	Soil DCGL 1.0 m (pCi/g)
Am-241	1.402E+02	3.053E+01
C-14	5.996E+01	1.019E+01
Ce-144	2.746E+02	2.319E+02
Cm-243	6.747E+01	3.060E+01
Cm-244	2.944E+02	5.766E+01
Co-58	3.631E+01	3.128E+01
Co-60	3.970E+00	3.086E+00
Cs-134	6.424E+00	4.237E+00
Cs-137	1.374E+01	7.656E+00
Eu-152	8.857E+00	7.748E+00
Eu-154	8.220E+00	7.168E+00
Eu-155	3.081E+02	3.027E+02
Fe-55	3.660E+04	2.122E+04
H-3	1.195E+04	8.655E+02
Ni-59	1.128E+04	2.307E+03
Ni-63	4.120E+03	8.424E+02
Np-237	4.723E+00	7.619E-01
Pu-238	1.752E+02	3.536E+01
Pu-239	1.578E+02	3.184E+01
Pu-240	1.578E+02	3.185E+01
Pu-241	5.666E+03	1.040E+03
Sb-125	2.662E+01	2.348E+01
Sr-90	1.111E+01	1.731E+00
Tc-99	1.356E+02	1.542E+01

Table F-5: Soil DCGL for ROC (Adjusted for IC Dose)

Radionuclide	Soil DCGL (pCi/g) 0.15 m	Soil DCGL (pCi/g) 1.0 m
C-14	5.70E+01	9.68E+00
Co-60	3.77E+00	2.93E+00
Cs-137	1.31E+01	7.27E+00
Eu-152	8.41E+00	7.36E+00

F.4 Dose Assessment Modeling for Groundwater

F.4.1 Early Decommissioning Project – Connecticut Yankee

As discussed above, the DCGLs for soil at CY were determined with the assumption that no groundwater contamination existed. In order to account for groundwater contamination present at the time of license termination (as CY 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 initial radionuclide concentration of 1 pCi/g (0.037 Bq/g) was used for the soil comprising the contaminated zone for each of the 20 radionuclides of interest for CY. This value does not affect the results of the calculation.

The RESRAD code is typically used to calculate radiation doses (and DCGLs) for a source above the water table. Per the verbal guidance from Argonne National Laboratory at meeting with the NRC, 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 were be 0 years (It should be noted that some of the values listed above, per recent Argonne Lab guidance, different parameters should be chosen as discussed in the next section).

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 F-2 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)

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

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

K_d = Distribution Coefficient of contaminated zone (cm3/g)

n = Total porosity of contaminated zone (Fraction)

Connecticut Yankee Groundwater DCGL Determination

To determine the Groundwater DCGLs at CY, the dose from all groundwater related pathways was determined for an assumed groundwater concentration of 10 pCi/Liter and then scaled to determine the concentration that corresponded to 25 mrem/yr. To determine the dose to be included in the compliance equation (Equation F-1) the individual DCGL concentrations that corresponded to 25 mrem/yr were scaled to the groundwater monitoring well sample results. The resulting Groundwater DCGL values are shown in Table F-6**.** Table F-6 also shows the EPA MCL concentrations (Reference 6-6) for most of the radionuclides of interest normally identified during power plant decommissioning. The last column of Table F-6 shows the calculated Total Effective Dose Equivalent (TEDE) dose that corresponds to the MCL concentrations for each radionuclide. These doses are calculated using the CY Groundwater DCGLs.

It should be noted these calculations are based on the situation at CY and other sites may have a different and site-specific analysis would be needed to support the selection of groundwater ROCs (and insignificant contributors to dose from the groundwater pathway), groundwater DCGLs and also the added risk from existing groundwater contamination. NRC guidance in DUWP–ISG-02 (Reference 5-6) provides detailed guidance on acceptable approaches for consideration of risk from existing groundwater contamination and methods to estimate potential exposure point concentrations if relying on monitoring data to assess risk.

Table F-6: CY Groundwater DCGLs

Radionuclide	Groundwater DCGL (pCi/L)	Maximum Contaminant Levels in Groundwater pCi/L	TEDE Dose Due to MCL Concentration in Groundwater mrem/yr
H-3	6.52 E+05	20,000	0.77
C-14	9.01 E+03	2,000	5.55
Mn-54	2.42 E+04	300	0.31
Fe-55	6.54 E+04	2,000	0.76
Co-60	1.14 E+03	100	2.19
Ni-63	3.15 E+04	50	0.04
Sr-90	2.51 E+02	8	0.80
Nb-94	6.75 E+03)	N/A	N/A
Tc-99	2.64 E+04	900	0.85
Ag-108m	4.24 E+03	N/A	N/A
Cs-134	3.42 E+02	80	5.84
Cs-137	4.31 E+02	200	11.6
Eu-152	7.33 E+03	200	0.68
Eu-154	5.05 E+03	60	0.30
Eu-155	3.25 E+04	N/A	N/A
Pu-238	1.51 E+01	N/A	N/A
Pu-239	1.36 E+01	N/A	N/A
Pu-241	4.60 E+02	N/A	N/A
Am-241	1.32 E+01	N/A	N/A
Cm-243	1.94 E+01	N/A	N/A

To determine the dose from the “existing groundwater” at CY, the monitoring well concentration which was within 100 meters (328 feet) of a survey area whose sample concentrations resulted in the highest calculated dose was used to show compliance with the NRC release criteria for that survey unit. The 100-meter distance was determined to be the maximum capture zone radius for a postulated water supply well drilled by a future resident farmer for the CY site. It should be noted that CY had installed an extensive monitoring well network, collected a great deal of hydrogeologic information and performed fate and transport modeling to support this use of this capture zone radius for assigning groundwater dose to individual survey units.

It should also be noted that there are a range of approaches for applying the dose from groundwater contamination. In using a graded approach, it may be practical in many cases to conservatively apply the maximum level of groundwater contamination across the site. In other cases, such as described for CY above, may be necessary to apply different dose values for individual survey areas/units. In this latter approach, additional data and detailed justification will be required to support the use of these different dose values.

F.4.2 Recent Guidance and Experiences in Determination of Dose from Groundwater

NRC Guidance on Calculating Groundwater Dependent Dose Conversion Factors

The following is a summary of guidance concerning this dose pathway from NRC guidance in Reference 5-6.

The DandD and the RESRAD-ONSITE conceptual models do not consider existing groundwater plumes located in the saturated zone when calculating dose or deriving screening values and DCGLs. In many cases, if there is existing groundwater contamination, licensees have apportioned a fraction of the dose limit to the groundwater pathway to demonstrate license termination rule criteria have been met (e.g., 5 mrem/yr (0.05 mSv/yr) of the 25 mrem/yr (0.25 mSv/yr) dose limit for unrestricted release). The RESRAD-ONSITE computer code has been used to calculate what is referred to as a pathway dose conversion factor (PDCF) for groundwater dependent pathways (e.g., drinking water, irrigation, livestock watering) in units of mrem/yr per pCi/L (or mSv/yr per Bq/L). The dose from existing groundwater contamination for all the potential uses of the groundwater are therefore considered. This section also discusses how RESRAD-ONSITE can be used to calculate the pathway dose conversion factors for a unit concentration of groundwater.

While the RESRAD-ONSITE computer code does not consider doses associated with existing groundwater plumes, the computer code can be used to calculate groundwater pathway PDCFs by running the code with some source concentration in the subsurface for an individual radionuclide and extracting the maximum groundwater well concentration and associated peak dose from groundwater dependent pathways such as drinking water, irrigation, fish ingestion, and livestock watering. Table F-7 below presents some example groundwater PDCF calculations using the default parameter values in RESRAD-ONSITE. Please be aware that these values are provided for illustration purposes only, and only site-specific parameter values should be used in calculating PDCFs for actual sites. For some constituents, the assumed distribution coefficient used in the simulation may be so high that the constituent is unable to travel to the saturated zone from the contaminated zone within the timeframe of the simulation and no PDCF can be calculated. In these cases, if site-specific conditions lead to the presence of the constituent in the saturated groundwater, then the conceptual model for contaminant release and transport may need to be revisited to ensure it aligns with dose modeling assumptions (e.g., presence of the source in the saturated zone, differences in geochemical conditions that resulted in faster transport rates). Adjustments to the model/parameters may be needed to enable calculation of PDCFs for those constituents. For example, updated versions after RESRAD-ONSITE version 6.5 allow placement of the source directly in the saturated zone. Placing the source in the saturated zone facilitates transfer of residual radioactivity to the well ensuring well concentrations will be realized to allow calculation of PDCFs.

It is important to note that while many of the site-specific (physical) parameters selected affect the ratio of concentration in groundwater per unit concentration in soil, they do not necessarily have an impact on the ratio of the dose per unit groundwater concentration. Only certain biosphere parameters influence the PDCFs (e.g., behavioral parameters such as drinking water intake, irrigation rates, and livestock water intake). It is always prudent to use the licensee's dose modeling files with their site-specific biosphere parameters already specified to calculate the pathway dose conversion factors. When performing these calculations, an analyst may need to adjust the graphics parameters, calculation times, or time integration points to get more accurate PDCFs. Checks on the stability of PDCF over different times and with different source parameter specifications (e.g., distribution coefficients and thickness of contaminated zone) should be made to ensure that rapidly depleting sources early in the simulation period do not lead to inaccuracies in the PDCF calculation. Refer to Reference 5-6 for more detail.

Table F-7: Example PDCFs

Radionuclide	Maximum GW Concentration from Unit Concentration	Peak Dose from Groundwater Dependent Pathways	PDCF mrem/yr per pCi/L[a]	Benchmark PDCF mrem/yr per pCi/L[a]
H-3	1065 pCi/L	0.10 mrem/yr @4.3 years	9.4E-05	4.4E-05[b]
Tc-99	1357 pCi/L	2 mrem/yr @4 years	1.5E-03	1.2E-03[b] 5E-03[c]
C-14	1367pCi/L	6.7 mrem/yr @4.4 years	4.9E-03	1.3E-03[b] 0.014[c]
Sr-90	0.01 pCi/L	6.9E-04 @227 years	0.069	0.34[c]

Note: RESRAD-ONSITE defaults were used except for 1 m cover and 0 m/yr erosion rate.

a. 1 mrem/yr = 0.01 mSv/yr; 1 pCi/L = 0.037 Bq/L [b] SRS F-Tank Farm Performance Assessment Pathway Dose Conversion Factors excluding stream pathways. [c] LaPlante, P.A. and K. Poor. 07/25/1997. "Information and Analyses to Support Selection of Critical Groups and Reference Biospheres for Yucca Mountain Exposure Scenarios." CNWRA 97-009. San Antonio, Texas: Center for Nuclear Waste Regulatory Analyses. August 1997. 20.05708.761, Q199709240001, ML040200056, Table E-1.

Fort Calhoun Approach to Dose Modeling for Existing Groundwater Contamination

The following is a summary of the information in the Fort Calhoun Station (FCS) LTP on this topic (Reference F-14).

The existing groundwater (GW) dose conversion factors (DCF) (mrem/yr per pCi/L) are derived from the results of the FCS embedded pipe Dose to Source Ratio (DSR) RESRAD analysis. The scenario used is immaterial as long as the contaminated zone area is at least 20,000 m^2 but the RESRAD results need to include non-negligible well water concentrations. The water dependent dose includes exposure from ingestion of drinking water, ingestion of plants subject to irrigation, and ingestion of meat and milk from cows that consume well water directly and consume fodder that is subject to irrigation. The existing GW DCFs are calculated in Excel using Equation F-3 and listed in Table F-8.

$$DCF_{egw,i} = \frac{D_{wd,t,i}}{C_{ww,t,i}} \qquad \textbf{Equation F-3}$$

where:
$DCF_{egw,i}$ = dose conversion factor for radionuclide i (mrem/yr per pCi/L)

$D_{wd,t,i}$ = water dependent dose at time t for radionuclide i (mrem/yr)
$C_{ww,t,i}$ = well water concentration at time t for radionuclide i (pCi/L)

Table F-8: Existing Groundwater Dose Conversion Factors for ROC

Radionuclide	Water Dependent Dose[1] at t=1 yr (mrem/yr)	Well Water Concentration[2] at t = 1 yr (pCi/L)	Dose Conversion Factor mrem/yr per pCi/L
C-14	0.45	1.663E+02	2.68E-03
Co-60	0.15	5.886E+00	2.52E-02
Cs-137	0.11	1.544E+00	6.86E-02
Eu-152	0.01	2.505E+00	3.63E-03
Sr-90	3.95	3.585E+01	1.10E-01

F.5 Adjusting Site Release Limits for Multiple Contaminated Mediums

As the site release criteria applies to all applicable dose pathways in the area being evaluated, the presence of multiple contaminated media needs to be considered. An example of this is where sample results for a site show contamination in groundwater (called "existing" groundwater) and contamination in the soil above the groundwater. In this case the current groundwater concentrations are normally compared to the groundwater DCGLs (or PDCF) and a yearly dose due to "existing" groundwater is calculated. The "existing" groundwater dose is then subtracted from the site release criteria (i.e., 25 mrem/year) to determine the dose that remains that can be applied to the soil in the unsaturated zone of the contaminated area. Appendix B illustrates how this allotment of dose was performed during the Connecticut Yankee Plant decommissioning.

F.6 Dose Assessment Modeling for Structures

The evolution of site release limits for structures was like that described for land areas in Chapter 6 and earlier in this appendix. The early U.S Nuclear Regulatory Commission (NRC) guidance primarily described one scenario: the Building Occupancy Scenario. Updated NRC guidance published in the 2003-2004 timeframe described the use of "realistic" scenarios. This section will first describe the Building Occupancy Scenario and then cover site-specific experiences with other scenarios.

This chapter contains a summary of the dose modeling experiences for concrete structures. More details on the dose calculation methods used at some early decommissioning sites is contained in EPRI Report, "Concrete Characterization and Dose Modeling During Plant Decommissioning: Detailed Experiences 1993 – 2007" (EPRI Report 1015502, Reference F-8) and other references given below.

F.6.1 Options for Developing DCGLs for Structures

Building Occupancy Scenario

The Building Occupancy Scenario is the default scenario for determining DCGLs for building surfaces. 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 formerly part of the plant. This room is assumed to have been released for use as an office. The DandD and RESRAD-Build computer codes are commonly used to calculate these building DGCLs.

NRC Published Screening Values for Structures

The NRC has published screening values for residual radioactivity in structures based on the Building Occupancy Scenario. These screening values are provided in NUREG-1757, "Consolidated Decommissioning Guidance," Volume 2, Rev. 2, Appendix H (Reference 1-4). These screening values were calculated by the DandD code using the conservative default input parameters. The use of the default parameters in the DandD 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 are as follows:

- Contamination on building surfaces should be surficial and not-volumetric [no more than 10 mm (0.39 in) of penetration]
- Residual Radioactivity on the building surface is mostly fixed (loose residual radioactivity no more than 10% of total surface activity)
- Use of the screening values on buried structures (e.g., drainage or sewer pipes) or equipment within the building requires justification and evaluation by the NRC on a case-by-case basis.

It should be noted that the screening values or Derived Concentration Guideline 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 1-4) 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. Additionally, it should be noted that revised versions of the DandD code exist, and that the latest version should be used by licenses if chosen to develop site specific DCGLs.

F.6.2 Dose Modeling Experience for Decommissioning Projects – Late 1990s to Early 2000s

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.

Connecticut Yankee Experience

Connecticut Yankee (CY) 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 a Subsurface Structure Dose Model (SSDM called BFM in their LTP). Each is described in more detail in the sections below.

CY Building Occupancy Scenario

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

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

Table F-9: CY Building Surface DCGLs (Building Occupancy Scenario) Compared to Generic Screening DCGLs

Radionuclide	CY Site Specific DCGL for Surface Sources (dpm/100 cm^2)	Generic Screening DCGL (dpm/100 cm2)	Ratio of Site Specific to Generic Screening DCGLs (Times Higher)
H-3	3.15 E+08	1.2 E+08	2.6
C-14	1.03 E+07	3.7 E+06	2.8
Mn-54	3.21 E+04	3.2 E+04	1
Fe-55	3.49 E+07	4.5 E+06	7.8
Co-60	1.11 E+04	7.1 E+03	1.6
Ni-63	3.60 E+07	1.8 E+06	20.0
Sr-90	1.27 E+05	8.7 E+03	14.6
Nb-94	1.71 E+04	Note 2	N/A
Tc-99	1.45 E+07	1.3 E+06	11.2
Ag-108m	1.65 E+04	Note 2	N/A
I-129	Note 1	3.5 E+04	N/A
Cs-134	1.65 E+04	Note 2	N/A
Cs-137	4.30 E+04	2.8 E+04	1.5
Eu-152	2.34 E+04	Note 2	N/A
Eu-154	2.19 E+04	Note 2	N/A
Eu-155	4.37 E+05	Note 2	N/A
Pu-238	4.87 E+03	Note 2	N/A
Pu-239	4.44 E+03	Note 2	N/A
Pu-241	2.29 E+05	Note 2	N/A
Am-241	4.27 E+03	Note 2	N/A
Cm-243	6.07 E+03	Note 2	N/A

Notes:
1. These radionuclides were determined not to be of concern at CY
2. Values for these radionuclides were not published in the Federal Register but can be determined by using the DandD dose modeling code.

Table F-9 lists the site-specific 25 mrem/yr Building Occupancy DCGLs for Connecticut Yankee, as well as the NRC Generic Screening DCGLs published in NUREG-1757, “Consolidated Decommissioning Guidance,” Volume 2, Rev. 2, Appendix H. Most of the site-specific values are significantly higher than the generic values. This illustrates the conservatism in the default input parameters used to derive the screening DCGLs.

CY 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-10: 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-10 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.

CY Subsurface Structures

CY used the Subsurface Structure Dose Model (SSDM, called BFM in their LTP) to determine the projected dose from the leaching of radionuclides from contaminated concrete. This calculated dose was the "Future Groundwater" dose component of the Equation F-1, the CY compliance equation. As discussed above, the "Future Groundwater" component might not apply in all situations. Connecticut Yankee submitted the SSDM for NRC approval as a "realistic" scenario.

Also, as discussed above, NRC encouraged the use of realistic scenarios, where appropriate, in its updated guidance in 2003. CY made the decision, in 2004, to dispose of all of the above ground concrete as either clean (containing no detectable plant-related contamination) or radioactive waste. This decision made the Concrete Debris Scenario no longer applicable. In the same timeframe, the Maine Yankee Decommissioning Project had received approval for a scenario using an SSDM. 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 SSDM 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-6) 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.

- In order to ensure that the ICI Sump would be considered inaccessible, the sump areas were filled with grout covering the area of activated concrete by at least 3 feet (0.9 m).

- Maine Yankee performed separate calculations for all of the various basements to remain on site. CY, as discussed above, assumed that all the activity would migrate to either the containment basement or to the discharge tunnels.

Use of the SSDM helped the plant avoid removal of the concrete from the Containment ICI Sump. It also avoided the projected need to completely remove the Spent Fuel Pool without the SSDM. The remediation needed behind the Fuel Pool liner was approximately 12 inches (30 cm) in the floor and essentially no remediation was needed in the walls. As the remediation of the Containment ICI Sump and the removal of the Spent Fuel Pool both would have been difficult remediation efforts, significant cost and schedule time were saved.

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 would 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.

Ranch Seco 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-10). RESRAD-BUILD Version 3.3 (Released Summer of 2005 by Argonne National Laboratory (ANL)) was chosen as the computational method to calculate structural surface DCGLs. RESRAD-BUILD as used at Rancho Seco considers seven exposure pathways:

- External exposure directly from the source,
- External exposure to materials deposited on the floor,
- External exposure due to air submersion,
- Inhalation of airborne radioactive particulates,
- Inhalation of tritiated water vapor,
- Inadvertent ingestion of radioactive material directly from the source, and
- Ingestion of materials deposited on the surfaces of the building compartments.

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

Rancho Seco Building Renovation/Demolition Scenario

The building renovation/demolition scenario, as described in NUREG/CR- 5512, Vol. 1 (Reference F-10) along with the input data template and input parameter values provided in ANL/EAD/03-1 (Reference F-12) specifies 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-11 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.

Rancho Seco 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-11 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.

It should be noted that NRC recent guidance contained in NUREG 1757 states that less likely but plausible scenarios be evaluated in addition to the most likely scenario(s). For the Rancho Seco scenarios discussed above, the most likely future scenario is that the impacted buildings are be demolished and the concrete rubble used as backfill for the on-site basements. Considering the revised NRC guidance, this future scenario would need to be analyzed to determine peak doses from the less likely but plausible exposure scenarios are significantly above the dose standard. If that were the case, the licensee would need to provide greater assurance that the exposure scenario is less likely to occur, especially during the period of unacceptably high dose.

Table F-11: Comparison of Rancho Seco Building Surface DCGLs for Alternate Scenarios

Radionuclide	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 building concrete 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.6.3 Recent Guidance and Experiences in Determination of Dose from Building Structures

Zion Dose Modeling – Concrete Debris

As with Connecticut Yankee discussed above, 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 5-2).

Although the concrete debris to remain onsite and used as clean fill can be viewed as having no dose impact, a dose value was 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 cm^2 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-12.

Table F-12: 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

Zion Experience with a Subsurface Structure Dose Model

The Zion site also utilized the SSDM to assess basements to remain after decommissioning in a similar fashion to CY. The SSDM 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
- Waste Water 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 SSDM 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 SSDM 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 backfill, 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 SSDM 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). SSDM Groundwater (GW) Dose Factors are then calculated for each Basement and each Radionuclide of Concern by combining the results of the two models with units of mrem/yr per mCi total inventory.

The SSDM also includes the dose from drilling spoils that are brought to the surface during the well installation, which is assumed to be at the time of maximum projected future groundwater concentrations. The drilling spoils are assumed to be comprised of fill material containing residual radioactivity at the maximum equilibrium concentrations. Any activity remaining in the concrete is also included in the drilling spoils source term. SSDM Drilling Spoils Dose Factors are also calculated in units of mrem/yr per mCi total inventory.

The final outputs of the SSDM are the Basement Derived Concentration Guideline Levels (DCGL), in units of pCi/m2, which are calculated using the SSDM GW and SSDM Drilling Spoils Dose Factors. The DCGLs for basement structure surfaces are calculated separately for the GW and Drilling Spoils scenarios and for the summation of both scenarios. The individual Basement Scenario DCGLs for structure surfaces are defined as “$DCGL_{BS}$” and represent a dose of 25 mrem/yr for each scenario individually. The basement summation DCGL for basement structure surfaces include the dose from both the GW and Drilling Spoils scenarios and represents a dose of 25 mrem/yr from both scenarios combined. The summation DCGL for basement structure surfaces is designated as the “$DCGL_{B}$” and is used during FSS to demonstrate compliance (Equivalent to the $DCGL_{W}$ as defined in MARSSIM). The DCGLs are radionuclide-specific concentrations that represent the 10 CFR 20.1402 dose criterion of 25 mrem/yr and are calculated for each ROC and each backfilled Basement.

Basement $DCGL_B$ 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 $DCGL_B$ calculations. Therefore, the $DCGL_B$ values calculated for the Turbine Basement also apply to the Circulating Water Discharge Tunnels and the $DCGL_B$ 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 SSDM Dose Factors and DCGLs. The Turbine Basement $DCGL_B$ values therefore also apply to the Steam Tunnel.

For more details on the dose modeling experiences for the Zion plant see Reference 5-2.

Fort Calhoun Experience with a Subsurface Structure Dose Model

The model for calculating dose and DCGLs from residual radioactivity in the backfilled basements at the Fort Calhoun site is named the Subsurface Structure Dose Model (SSDM, called BFM in their LTP). The basements to remain are Containment Building, Auxiliary Building, Turbine Building, Intake Structure, and Circulating Water Tunnels. There are two media in backfilled basements: walls/floors and embedded pipe. A separate DCGL is calculated for each media. Detailed descriptions of the source term abstractions, conceptual models, and DCGL calculations for each media are provided the Fort Calhoun LTP (Reference 6-10). A brief overview is provided here.

All basements include contaminated walls and floors (walls/floors) as a media or source term. The walls/floors are comprised of concrete and the remaining steel liner in Containment. Penetrations are included with the wall/floor media and have the same DCGLs. A single DCGL is calculated that applies to the Auxiliary Building, Turbine Building, Intake Structure, and Circulating Water Tunnel basements. A separate DCGL is required for Containment Building due to the presence of the liner.

The SSDM wall/floor model includes three source release pathways:

- instant release from concrete (or from the surface of the Containment liner) to the water in the pore space of the fill material,
- capture of concrete (or steel liner in Containment) with drilling spoils generated during the installation of an onsite well, and
- excavation of concrete walls (or steel liner in Containment).

The doses from the three wall/floor release pathways are summed to calculate the final wall/floor DCGL.

The Auxiliary and Turbine Building basements also contain embedded pipe. The walls/floors and the embedded pipe have significant differences in source terms, physical configurations, conceptual models, and source term abstraction and release. Therefore, the DCGL calculation methods and results for walls/floors and embedded pipe are described separately later in this appendix. The dose from walls/floors and embedded pipe are treated as separate media when calculating the final dose for FCS compliance with the 25 mrem/yr dose criterion.

Fort Calhoun SSDM Wall/Floor Scenarios

There are three scenarios that result in dose to the AMCG from residual radioactivity in the walls/floors: in situ scenario, excavation scenario, and drilling spoils scenario. The doses from the three scenarios are summed to determine the final wall/floor DCGL. This section describes the source term configuration, conceptual model, source term abstraction, and mathematical model for each of the three SSDM wall/floor scenarios. The dose assessment methods and the DCGL calculations for each scenario are described in more detail in the Fort Calhoun LTP.

Fort Calhoun SSDM Wall/Floor Source Term Configuration and Spatial Variability

This section discusses the source term in basement walls/floors (including the steel liner to remain in Containment).

For Fort Calhoun, the major basement source term at license termination is expected to be in the Auxiliary Building basement concrete. Minimal contamination was introduced into the Turbine Building and Circulating Water Tunnels during operation or is expected to be introduced during decommissioning. The operations of the Intake Structure did not involve contact with plant-derived radionuclides. All concrete that is inside of the steel liner will be removed from the Containment Building leaving only the steel liner and the concrete outside of the liner. The vast majority of the contamination in the Containment Building will be removed with the concrete. The source term configuration and spatial variability in each basement is discussed below in a level of detail commensurate with the contamination potential at the time of license termination.

<u>Auxiliary Building Basement</u>

The Auxiliary Building contained all support systems for reactor operations that are not located in the Containment Building. In addition to reactor support systems typical of a pressurized water reactor (PWR) Auxiliary Building, the structure also contains the spent fuel pool, spent fuel support systems, emergency diesel generators, emergency core cooling and shutdown cooling systems. The Auxiliary Building is a large and complex structure with a wide range of radiological conditions within the structure.

Concrete wall/floor core samples were collected in the Auxiliary Building basement during characterization. Consideration was given to locations that exhibited measurable radioactivity (identified during the scan survey), depressions, discolored areas, cracks, low point gravity drain points, and actual or potential spill locations. The detailed core sample results are provided in the Chapter 2 of the Fort Calhoun LTP. All Auxiliary Building basement interior walls will be removed leaving only the 971 and 989 foot elevation floors and outer walls at license termination. Therefore, only the samples from the 971 and 989 foot elevation floors and outer walls of the Auxiliary Building basement are germane to this discussion of source term present at the time of license termination.

The operational DCGL (OpDCGL) for Fort Calhoun is based on Cs-137 concentration only and assumes a wall/floor Base Case DCGL (BcDCGL) of 6.90E+06 pCi/m^2 for Cs-137 which corresponds to 245 pCi/g uniformly distributed in a 0.5-inch layer of concrete. The OpDCGL a priori fraction is assumed to be 0.15 resulting in 37 pCi/g Cs-137. The concrete core results for the 0.5 in -1.5 in depths indicated that concrete remediation deeper than 0.5 inch is, or may be, necessary.

From a review of the initial characterization sample results, two observations were made regarding the Auxiliary Building basement source term configuration and spatial variability:

- The general area dose rates are much lower than the maximum dose rates identified in a given room. This indicates that the elevated contamination may be more localized than widespread.

- There are a number of rooms where significant contamination is found deeper than 0.5 inch in the concrete. However, sampling was limited to a depth of 1.5 inch at most locations. A further investigation was planned to bound the area and depth in the elevated areas.

The predominate radionuclide identified is Cs-137. Co-60 and Cs-134 are also positively identified but at much lower concentrations than Cs-137. The average Co-60 concentration in the 971 foot elevation was 0.3% of the Cs-137 concentration. For the 989 foot elevation, the average Co-60 concentration is 1.4% of the Cs-137 value. The Cs-134 fractions are lower than Co-60.

Eleven samples from the Auxiliary Building basement with elevated Cs-137 concentrations were sent to GEL laboratories for analysis of non-gamma emitting radionuclides (commonly referred to as hard to detect (HTD) radionuclides). Positively identified HTD radionuclides include H-3, C-14, Ni-59, Ni-63, Tc-99 and Sr-90. The HTD radionuclide with the highest abundance relative to Cs-137 is Ni-63.

Containment Building Basement

Significant concentrations of activated concrete were identified during the characterization of the under-vessel area (977 foot elevation). Lower concentrations were identified in the concrete at the 994 foot elevation with very limited indication of concrete activation. See Chapter 2 of the Fort Calhoun LTP for the sample results. However, all of the concrete in the Containment Building was removed down to the steel liner. The source term at the time of FSS and license termination was postulated to be limited to surface contamination on the liner at low levels due to cross-contamination from dust generated during concrete removal. Nine concrete core samples were collected under the liner with no results above the minimum detectable activity (MDA).

Turbine Building Basement, Circulating Water Tunnels, and Intake Structure

From operational history, minimal contamination was expected in the Turbine Building basement, Circulating Water Tunnels, and Intake Structure. The liquid radwaste discharge point is within the Circulating Water Tunnels but the low concentrations in the liquid combined with high flow rate through the tunnels were expected to minimize the extent of concrete contamination. Eighteen concrete core samples were collected from the Turbine Building basement during characterization and six cores were collected from the Intake Structure. No cores were collected from the Circulating Water Tunnels due to access issues. There were no plant-derived radionuclides detected in any of the concrete core samples. Although there is always the potential for cross contamination during demolition and decommissioning activities, minimal residual radioactivity is expected to be present at the time of license termination.

Fort Calhoun SSDM Wall/Floor Conceptual Model, Source Term Abstraction and Release

The basements of the Turbine Building, Containment Building, Auxiliary Building, Circulating Water Tunnels, and Intake Structure will remain with all interior walls removed. One exception is the Turbine Building where the pedestals will remain within the interior of the basement. The basements will be demolished to at least 3 feet (0.92 m) below grade which corresponds to an elevation of 1001 foot Above Mean Sea Level (AMSL).

After remediation and FSS are completed, contingent upon the completion of confirmatory surveys and regulatory approval, the Turbine Building, Containment Building, and Auxiliary Building basements will be backfilled to the original grade of 1004 feet AMSL. The Circulating Water Tunnels and Intake Structure will be backfilled with flowable fill (grout) or fill material from the rail spur expansion. All basement walls/floors will therefore have a minimum cover thickness of 0.92 m.

The SSDM wall/floor source term includes surface and volumetric contamination on and within the basement concrete and surface contamination on the remaining steel liner in Containment. The concrete and remaining sleeves in the penetrations are assumed to be a part of the wall/floor source term and have the same DCGLs as walls/floors. There are three conceptual model scenarios that have different source term release mechanisms. The doses from each of the three scenarios are summed to determine the SSDM wall/floor DCGL. The scenarios and release mechanisms are:

- in situ scenario: release from concrete (or from the surface of the Containment liner) to the water in the pore space of the fill material,
- drilling spoils scenario: incorporation of concrete into drilling spoils generated during the installation of an onsite well, and
- excavation scenario: excavation of basement concrete walls and steel liner of Containment.

The environmental pathways and exposure pathways that apply to each of the three scenarios are listed in Table 6-5, source numbers 13, 14, and 15.

The in situ scenario source term release pathway from concrete and the steel liner leads to the contamination of fill material and water in the pore space of the fill. As discussed above, the majority of residual radioactivity is expected to be in the concrete as opposed to the liner. The ratio of the concentration in fill to that in the water is a function of the radionuclide specific K_ds.

The source term release pathway for the drilling spoils scenario is the removal of contaminated concrete during the installation of a water well. The contaminated concrete is brought to the ground surface with the overlying fill material and uniformly mixed within the drilling spoils.

The third release pathway is the excavation of basement concrete walls and the steel liner from Containment. The contaminated and uncontaminated portions of concrete are inadvertently mixed during the process of excavation, rebar removal, sizing, and spreading the concrete debris on the ground with a thickness of 1 m. The excavated concrete is treated as soil in the dose calculation. The activity on the steel liner is released to the surface soil after excavation. The excavation of the liner would be a major construction project with significant disturbance of the surrounding soil. The released activity could be mixed over a range of depths in soil. Therefore, both a 0.15 m and 1.0 m mixing depth are evaluated and the maximum dose is used for the DCGL calculation.

The SSDM conceptual model and source term configuration includes several abstractions and simplifications. The source is assumed to be a volumetric or surface layer at uniform concentrations (pCi/m^2) in or on the walls and floors of all basements. The assumption of uniform distribution over all surfaces overestimates the source term because, as discussed in above, all basements except the

Auxiliary Building basement are expected to contain minimal residual radioactivity at license termination. The thickness of the source in concrete is immaterial because all contamination at all depths is assumed to instantly release to the adjacent fill immediately after license termination. Instant release is a conservative assumption because the residual radioactivity would actually release more slowly by diffusion from concrete allowing for radioactive decay as well as continual source depletion as the water containing radioactivity is removed by the well. For the drilling spoils and excavation scenarios, the contamination is assumed to remain in the concrete which maximizes the source term.

The water supply well in the SSDM in situ scenario is assumed to be located 1 m from the wall. To reach the well, the residual radioactivity in the concrete is assumed to instantly release and uniformly mix in the fill between the wall and the well. A 1 m mix distance corresponds to a surface area to volume ratio (SA/V) of 1, i.e., the residual radioactivity contained in 1 m^2 of wall surface is mixed into 1 m^3 of fill. The concentration in the fill is an inverse linear function of the distance from the wall to the well and the corresponding mixing distance. A reasonable alternate assumption would be to locate the well in the center of a given basement.

The structures at Fort Calhoun were assumed to provide no obstruction to groundwater flow, i.e., are not present. The fill is assumed to have the characteristics of the soil in the saturated zone, i.e., sand. The actual fill material slated to be used is a silt loam soil type. The K_ds of silt loam are higher than sand K_ds and would therefore result in lower well water concentrations in the SSDM in situ scenario and correspondingly lower dose. But the characteristics of the fill is not the only consideration. Under the assumption that the walls are not present and that the immediately adjacent surrounding soil is saturated zone sand, the water flowing from the fill to the surrounding sand would equilibrate in accordance with the sand K_d. If the well is located outside of the basement at the downstream edge, the well water concentrations may not differ greatly between sand or silt loam fill material.

The location of the well, and the corresponding mixing distance in fill, and the assumption of flow or no flow groundwater conditions in the basements are important considerations in the SSDM *in situ* conceptual model. The source term represented by the volume of contaminated fill in all basements, after release and uniform mixing (total wall/floor surface area multiplied by the 1 m mix distance), as well as the assumption of groundwater flow conditions were used in RESRAD to calculate dose to the AMCG.

The conceptual model for the SSDM excavation scenario includes the excavation of at least the top 2 m of concrete walls (or steel liner in Containment) which is approximately 3 m bgs. The minimum wall thickness in any basement (2 ft) is used and the concrete is assumed to be spread over a 1 m depth on the ground surface. A typical excavation process for a backfilled structure would entail using a medium sized excavator with a 1.0 to 1.5 cubic yard bucket to excavate and stockpile fill. After removing the fill to the planned excavation depth, a hoe-ram would be used to pound out the concrete walls. The concrete would be segregated, the rebar removed, and remaining concrete size reduced. The SSDM excavation scenario assumes that the size reduced concrete is used as onsite fill.

The SSDM excavation scenario applies to concrete. A check calculation was performed assuming that the activity is released to the fill as opposed to remaining in the concrete. The concentrations in the excavated fill were shown to be less than the concentrations in the excavated concrete.

Fort Calhoun SSDM In situ Scenario Mathematical Model

RESRAD was used to calculate DSRs (mrem/yr per pCi/g) for the SSDM in situ scenario. There are three conceptual model assumptions that require additional discussion to demonstrate that the RESRAD non-dispersion model is applicable and conservative for the in situ scenario: 1) source term geometry after release and mixing in fill, 2) assuming all walls/floors are in the saturated zone, and 3) assuming groundwater flow through the basements. After confirming that these three assumptions can be conservatively applied in RESRAD, the converted source term (after release from concrete and mixing in fill) was used in a standard manner in RESRAD to calculate dose from all exposure pathways in the resident farmer scenario.

As discussed above, the well location and mixing distance are correlated and create the source term geometry. Instant release and mixing are conservatively assumed. After release from the wall by diffusion, the contamination moves to the well in accordance with the natural gradient as augmented by the gradient induced by the well pumping. Any mixing time other than instant release will lower the concentration in the well.

Fort Calhoun Source Term Geometry

The source term geometry in fill after release from walls is a function of the well location and corresponding mixing distance in fill. Assuming the well is located 1 m from the wall, the source term geometry is a 1 m thick layer of contaminated fill adjacent to all walls. The mixing distance from the floors is also assumed to be 1 m for consistency but, as discussed below, the well water concentration from floor contamination is not sensitive to the mixing distance.

The RESRAD non-dispersion groundwater model was used to calculate well water concentrations for the SSDM in situ scenario. However, the geometry cannot be modeled directly in RESRAD which requires a single contiguous source with a given area and depth. As a simplifying assumption, a source term is constructed for use in RESRAD which has a volume equal to the total surface area of all walls and floors in all basements multiplied by the assumed 1 m mix distance in fill. This represents the conceptual volume of contaminated fill after release from all basement surfaces and mixing into a 1 m fill thickness (i.e., SA/V of 1).

The total basement surface area is 16,700 m^2 which corresponds to a source volume of 16,700 m^3. A nominal source term thickness of 4 m is assumed which is the approximate height of basement walls at the 989 foot elevation Auxiliary Building floor and 990 foot elevation of the Turbine Building floor. The conceptual source is assumed to be 60 m wide (the approximate width of the Auxiliary and Turbine Building basements) and the length parallel to aquifer flow is assumed to be 70 m (16,700 m^3/ (60 m x 4 m)). The well intake depth is set to the same value as the source thickness, i.e., 4 m.

The primary performance objective of the conceptual model source term configuration is to ensure that the well water concentration calculated by the RESRAD non-dispersion model is at the theoretical maximum. This is accomplished with the 16,700 m^2 area and 4 m depth as discussed below and demonstrated using Equation F-4. But it should be noted that this could also be accomplished with essentially any conceptual source term configuration if the well depth is equal to or less than the conceptual source term depth.

The entire source term is assumed to be in the saturated zone. RESRAD ONSITE Version 7.2 allows the source to be entirely submerged in the saturated zone but the use of RESRAD ONSITE Version 7.2 under saturated conditions requires justification according to the NRC. The performance of the RESRAD non-dispersion model with a fully submerged source term was tested. The code was run with Cs-137 and Sr-90 using a 16,700 m^2, 4 m deep, fully submerged source term with a length parallel to flow of 70 m. The well intake depth is set to the same value as the source thickness, i.e., 4 m. The remaining RESRAD parameters were set to those used to determine the Fort Calhoun soil DCGLs with a few modifications that are discussed in the FCS LTP**.** The well water concentrations calculated by RESRAD were compared to the theoretical maximum concentrations in the fill pore space water, after instant release and mixing, using Equation F-4**.** Achieving a well water concentration that is equal to the theoretical maximum in the fill pore space water is considered a conservative and acceptable result.

To determine the theoretical maximum water concentrations, Equation F-4 was used with a unit concentration of 1 pCi/g in the fill, and K_ds of 158 and 6.6, for Cs-137 and Sr-90, respectively (25^{th} percentiles of saturated zone sand K_d PDFs). A unit source volume (fill) of 1 m^3 was assumed for the calculation along with the saturated zone.

$$C = \frac{I}{[V(\theta + \rho Kd)]}$$ **Equation F-4**

where:

C= concentration in water (pCi/L)
I = inventory (pCi) in fill = 1.49×10^6 pCi
V = mixing volume (L) = 1000 L
Θ = total porosity of the saturated zone = 0.45
ρ = bulk density (g/cm^3) = 1.49 g/cm^3
K_d = distribution coefficient (cm^3/g)

and,

$$I = V_f 1x10^6 \rho_f C_f$$

where:

$V_{f\,=}$ volume of fill = 1 m^3
$1x10^6$ = conversion factor cm^3/m^3
ρ_f = bulk density of fill = 1.49 g/m^3
C_f = unit concentration in fill = 1 pCi/g

The theoretical maximum pore space water concentrations calculated by Equation F-4 are 6.32 pCi/L and 144.9 pCi/L for Cs-137 and Sr-90, respectively. The well water concentrations calculated by RESRAD at time = 0 are the same at 6.23 pCi/l and 144.9 pCi/L, respectively. The performance of the RESRAD non-dispersion model, when the source term is fully submerged, is therefore considered conservative and acceptable. In addition, a sensitivity analysis was performed to confirm that the RESRAD calculated well water concentrations, when the source term is fully submerged, are not sensitive to well pumping rate. The concentrations remained at the theoretical maximum values over a wide range of well pumping rates.

A second assessment evaluated whether the maximum water concentration in fill, as calculated by RESRAD, assuming instant release and a 1 m mix distance (SA/V=1), is conservative when compared to the well water concentrations calculated using the actual source term geometry. Groundwater Vistas is a more complex groundwater modeling package that is used to directly evaluate the actual source term geometry. Well water concentrations calculated by RESRAD, using the conceptual model source term described above, and by Groundwater Vistas using the actual source term geometry were compared. The 3-D groundwater flow modeling was performed using MODFLOW-2005, particle tracing was calculated using MODPATH version 7, and radionuclide transport was completed using MT3DMS. A description of how these models were used is contained in the FCS LTP.

The results of the comparison of the MT3DMS and RESRAD model results were that the MT3DMS groundwater modeling of a more realistic basement source term geometry confirms that the well water concentrations calculated using RESRAD with a source term comprised of a contiguous volume that is 16,700 m^2 in area and 4 m deep, and is fully submerged in the saturated zone with a 4 m well pump intake depth, will provide conservative well water concentrations. Coupling the MT3DMS results with the assumptions that the well is located at a distance of 1 m from the wall that the activity is instantly released, and that the well is installed immediately after license termination, provides high confidence that the SSDM in situ scenario conceptual model will provide well water concentrations that are less than would actually be encountered by future site occupants.

More details on this analysis and its results are contained in the Fort Calhoun LTP.

Fort Calhoun SSDM In situ Scenario RESRAD Uncertainty Analysis

The process used for the uncertainty analysis is the same as used for the Fort Calhoun Soil DCGLs with the exception that the number of observations for certain actinides were reduced to 300 due to long run times. The parameters used for the SSDM in situ scenario uncertainty analyses are the same as those used for soil with changes that are noted in the Fort Calhoun LTP.

Fort Calhoun SSDM In situ Scenario Uncertainty Analysis Results and Deterministic Parameter Selection

The results of the uncertainty analysis and the selected deterministic parameters are provided in Table F-13 and Table F-14 to illustrate how input parameters were selected based on these results.

Table F-13: Ft. Calhoun SSDM Wall/Floor in situ Scenario RESRAD Parameter Uncertainty Analysis Results for Non-Nuclide Specific Parameters which were Determined to be Sensitive and Selected Deterministic Values

Parameter	Correlation to Dose	Radionuclide	Basis of Deterministic Parameter Selection	Selected Deterministic Value
Cover erosion rate (m/y)	positive	Am-241, C-14, Eu-152, Ni-59, Pu-239, Pu-240	75^{th} percentile	2.92E-03
Evapotranspiration coefficient (unitless)	positive	H-3, Tc-99	75^{th} percentile	0.87
Runoff coefficient (unitless)	positive	H-3, Tc-99[2]	75^{th} percentile	0.63
Depth of roots (m)	positive	Am-241, C-14, Ce-144, Cm-243, Cm-244, Cs-134, Cs-137, Co-58, Co-60, Eu-152, Eu-154, Eu-155, Fe-55, H-3, Ni-	75^{th} percentile	3.08

Parameter	Correlation to Dose	Radionuclide	Basis of Deterministic Parameter Selection	Selected Deterministic Value
		59, Ni-63, Np-237, Pu-238, Pu-239, Pu-240, Pu-241, Sb-125, Sr-90, Tc-99		
Wet foliar interception fraction of leafy vegetables (unitless)	positive	Ce-144, Fe-55, Np-237, Pu-241[2], Pu-239[2], Pu-240	75th percentile	0.70
Weathering removal constant all vegetation (unitless)	negative	Ce-144, Fe-55, Ni-59[2], Ni-63[2], Np-237, Pu-238, Pu-239, Pu-240, Pu-241, Tc-99	25th percentile	21.5
Wet weight crop yield of fruit, grain, and non-leafy vegetable (kg/m^2)	negative	Ce-144, Fe-55, Np-237[2], Pu-241[2], Pu-239[2], Pu-240	25th percentile	1.27

1) not sensitive (NS)

2) PRCC is less than |0.25| but correlation is indicated. Confirmed to have slight correlation to dose by single parameter sensitivity analysis.

Table F-14: Ft. Calhoun SSDM Wall/Floor in situ Scenario RESRAD Parameter Uncertainty Analysis Results for K_d of Contaminated Zone and Unsaturated Zone and Selected Deterministic Values

Nuclide	Correlation to Dose	Basis of Deterministic Parameter Selection	Selected Deterministic Value (cm^3/g)
Am-241	negative	25^{th} percentile	2.69E+02
C-14	negative	25^{th} percentile	1.26E+00
Ce-144	NS	50^{th} percentile	3.99E+02
Cm-243	negative	25^{th} percentile	5.72E+02
Cm-244	negative	25^{th} percentile	5.72E+02
Co-58	negative	25^{th} percentile	3.70E+01
Co-60	negative	25^{th} percentile	3.70E+01
Cs-134	negative	25^{th} percentile	1.58E+02
Cs-137	negative	25^{th} percentile	1.58E+02
Eu-152	negative	25^{th} percentile	9.45E+01
Eu-154	negative	25^{th} percentile	9.45E+01
Eu-155	negative	25^{th} percentile	9.45E+01
Fe-55	NS	50^{th} percentile	3.21E+02
H-3	negative	25^{th} percentile	4.30E-02
Ni-59	negative	25^{th} percentile	2.76E+01
Ni-63	negative	25^{th} percentile	2.76E+01
Np-237	negative	25^{th} percentile	5.49E+00
Pu-238	negative	25^{th} percentile	1.56E+02
Pu-239	negative	25^{th} percentile	1.56E+02
Pu-240	negative	25^{th} percentile	1.56E+02
Pu-241	negative	25^{th} percentile	1.56E+02
Sb-125	negative	25^{th} percentile	5.07E+00
Sr-90	negative	25^{th} percentile	6.57E+00
Tc-99	negative	25^{th} percentile	1.90E-02

1) not sensitive (NS)

Fort Calhoun SSDM In situ Scenario Initial Suite DCGL

The SSDM in situ scenario DCGL ($DCGL_i$) is calculated in two steps. First, the unit concentration of activity in the fill (pCi/g per pCi/m^2) is calculated in Excel using Equation F-5. The well location (mixing distance in fill) was assumed to be 1 m from the walls and floors. The activity in a 1 m^2 area of the wall instantly releases and mixes with a 1 m^3 volume of fill. The $DCGL_i$ are calculated using the unit concentration from Equation F-5 and the DSRs calculated with the parameters developed as discussed in the last section**.** The $DCGL_i$ values, not corrected for IC dose ae listed in Table F-15.

$$C_{f,u} = \frac{A_{c,u}}{V_{f,u} 1x10^6 \rho_f}$$ **Equation F-5**

where:

$C_{f,u}$ = unit concentration in fill (pCi/g per pCi/m^2)
$A_{c,u}$ = unit activity of 1 pCi over a 1 m^2 area of concrete (pCi)
$V_{f,u}$ = unit fill volume of 1 m^3
Conversion Factor = $1x10^6$ cm^3 per m^3
r_f = bulk density of fill (assumed to be sand)

$$DCGL_{i,j} = \frac{25}{DSR_{i,j} C_{f,u}}$$ **Equation F-6**

where:

$DCGL_{i,j}$ = in situ scenario DCGL for radionuclide j (pCi/m^2)
$C_{f,u}$ = unit concentration in fill (pCi/g per pCi/m^2) from **Error! Reference source not found.**F-5.
$DSR_{i,j}$ = SSDM in situ dose to source ratio for radionuclide j (mrem/yr per pCi/g)
25 = 25 mrem/yr dose criterion

Table F-15: Ft Calhoun SSDM in situ Scenario Initial Suite DSRs and DCGLs (No IC Dose Correction)

Radionuclide	DSR (mrem/yr per pCi/g)	$DCGL_i$ (pCi/m^2)
Am-241	8.025E+00	4.642E+06
C-14	1.591E+00	2.341E+07
Ce-144	2.441E-02	1.526E+09
Cm-243	2.800E+00	1.330E+07
Cm-244	2.238E+00	1.664E+07
Co-58	5.019E-02	7.422E+08
Co-60	1.299E+00	2.868E+07
Cs-134	1.854E+00	2.009E+07
Cs-137	1.472E+00	2.531E+07
Eu-152	4.061E-02	9.173E+08
Eu-154	5.906E-02	6.307E+08
Eu-155	9.177E-03	4.059E+09
Fe-55	1.382E-03	2.695E+10
H-3	1.366E-01	2.727E+08
Ni-59	1.852E-02	2.011E+09
Ni-63	5.072E-02	7.344E+08
Np-237	4.406E+02	8.454E+04
Pu-238	1.173E+01	3.176E+06
Pu-239	1.302E+01	2.861E+06
Pu-240	1.302E+01	2.861E+06
Pu-241	2.577E-01	1.445E+08
Sb-125	3.042E-01	1.225E+08
Sr-90	2.694E+01	1.383E+06

Radionuclide	DSR (mrem/yr per pCi/g)	$DCGL_i$ (pCi/m^2)
Tc-99	4.512E+00	8.256E+06

Ft Calhoun SSDM Drilling Spoils Scenario Mathematical Model

The mathematical model for the drilling spoils scenario treats the spoils as soil after spreading over a 0.15 m depth on the ground surface and then uses RESRAD to calculate the DCGL.

The diameter of the drill borehole was assumed to be 8 inches (0.203 m). The minimum depth to a floor in the Auxiliary Building, Turbine Building, or Containment Building basement is 8.5 feet (2.59 m) which is the distance from the ground surface (1004 foot elevation) to the 995.5 foot elevation of the spent fuel pit floor in the Auxiliary basement. The minimum distance maximizes the radionuclide concentration in the drilling spoils. The drill is assumed to travel 1 inch (0.0254 m) into the floor before drilling is stopped due to meeting refusal from the concrete. The total drilling depth, including fill and concrete, is therefore 2.62 m. The spoils are assumed to be spread on the ground at a thickness of 0.15 m. The area of the drilling spoils after spreading is the volume divided by 0.15 m.

The equations and parameters used to calculate the DSR and DCGLs for the drilling spoils scenario are listed in the Fort Calhoun LTP. The drilling spoils DSRs and DCGLs ($DCGL_{ds}$), with no IC dose correction, are listed in Table F-16.

Table F-16: Ft. Calhoun SSDM Drilling Spoils Initial Suite DSRs and Base Case DCGLs

Radio-nuclide	DSR (mrem/yr per pCi/g)	$DCGL_{ds}$ (pCi/m^2)
Am-241	3.627E-03	2.687E+10
C-14	6.142E-07	1.587E+14
Ce-144	4.556E-03	2.139E+10
Cm-243	1.636E-02	5.957E+09
Cm-244	1.169E-03	8.336E+10
Co-58	3.390E-02	2.875E+09
Co-60	2.902E-01	3.358E+08
Cs-134	1.698E-01	5.739E+08
Cs-137	7.161E-02	1.361E+09
Eu-152	1.371E-01	7.108E+08
Eu-154	1.460E-01	6.675E+08
Eu-155	5.241E-03	1.859E+10

Radio-nuclide	DSR (mrem/yr per pCi/g)	$DCGL_{ds}$ (pCi/m^2)
Fe-55	4.148E-08	2.349E+15
H-3	8.452E-07	1.153E+14
Ni-59	2.433E-07	4.005E+14
Ni-63	6.613E-07	1.474E+14
Np-237	3.059E-02	3.186E+09
Pu-238	1.874E-03	5.200E+10
Pu-239	2.061E-03	4.728E+10
Pu-240	2.058E-03	4.735E+10
Pu-241	8.992E-05	1.084E+12
Sb-125	4.713E-02	2.068E+09
Sr-90	1.380E-03	7.062E+10
Tc-99	9.713E-05	1.003E+12

Ft. Calhoun SSDM Excavation Scenario Mathematical Model

As discussed above, the SSDM excavation scenario conceptual model assumes that the wall with the minimum thickness is excavated and spread over a 1 m depth on the ground surface. Mixing of the contaminated and uncontaminated portions of the concrete occurs during excavation, removing rebar, and sizing the concrete to allow for use as onsite backfill. The minimum wall thickness for all basements is 2 ft. The concrete is conservatively assumed to have the characteristics of soil. The soil DCGLs were used to calculate dose from excavated concrete and corresponding concrete DCGLs in units of pCi/m^2.

The surface area of the excavated concrete, after spreading on the ground surface, was conservatively assumed to be large such that area factors do not apply.

The equations and parameters used to calculate the DSR and DCGLs for the excavation scenario are listed in the Fort Calhoun LTP.

The DCGL for the excavated steel liner ($DCGL_{el}$) requires a different mathematical model than that used for concrete. Mixing within the volume of the steel liner would not occur during excavation. After excavation and placement on the ground surface, the activity on the liner surface is assumed to instantly release and mix with the underlying soil over an area equal to the surface area of the liner. The excavated liner surface area, and corresponding underlying soil area is assumed to be large such that area factors do not apply. The mixing depth in soil after release from the liner may be less than 1 m. Therefore, a sensitivity analysis was performed to evaluate the $DCGL_{el}$ with soil mixing depths of 0.15 m and 1 m.

A unitized activity of 1 pCi over a 1 m^2 area of the excavated liner is assumed to mix within the underlying 1 m^2 area of soil at each depth (0.15 m or 1 m). The resulting concrete excavation DCGLs (not corrected for IC dose) are shown in Table F-17.

Table F-17: Ft. Calhoun Concrete Excavation DCGLs ($DCGL_{ec}$) (No IC Dose Correction)

Radionuclide	$DCGL_{ec}$ (pCi/m^2)	Radionuclide	$DCGL_{ec}$ (pCi/m^2)
Am-241	4.094E+07	Fe-55	2.846E+10
C-14	1.367E+07	H-3	1.161E+09
Ce-144	3.110E+08	Ni-59	3.094E+09
Cm-243	4.104E+07	Ni-63	1.130E+09
Cm-244	7.733E+07	Np-237	1.022E+06
Co-58	4.195E+07	Pu-238	4.742E+07
Co-60	4.139E+06	Pu-239	4.270E+07
Cs-134	5.682E+06	Pu-240	4.271E+07
Cs-137	1.027E+07	Pu-241	1.395E+09
Eu-152	1.039E+07	Sb-125	3.149E+07
Eu-154	9.613E+06	Sr-90	2.321E+06
Eu-155	4.060E+08	Tc-99	2.068E+07

The steel liner excavation scenario DCGL ($DCGL_{el}$) calculations and sensitivity analyses are performed in Excel for both the 0.15 and 1 m mixing thickness. The 0.15 m soil mixing thickness resulted in the minimum $DCGL_{el}$ for all radionuclides and were therefore assigned as listed in Table F-18 (not corrected for IC dose).

Table F-18: Ft. Calhoun Liner Excavation DCGLs ($DCGL_{el}$) (No IC Dose Correction)

Radionuclide	$DCGL_{el}$ (pCi/m^2)	Radionuclide	$DCGL_{el}$ (pCi/m^2)
Am-241	3.155E+07	Fe-55	8.235E+09
C-14	1.349E+07	H-3	1.298E+09

Radionuclide	$DCGL_{el}$ (pCi/m^2)
Ce-144	6.179E+07
Cm-243	1.518E+07
Cm-244	6.624E+07
Co-58	8.170E+06
Co-60	8.933E+05
Cs-134	1.445E+06
Cs-137	3.092E+06
Eu-152	1.993E+06
Eu-154	1.850E+06
Eu-155	6.932E+07

Radionuclide	$DCGL_{el}$ (pCi/m^2)
Ni-59	2.538E+09
Ni-63	9.270E+08
Np-237	1.063E+06
Pu-238	3.942E+07
Pu-239	3.551E+07
Pu-240	3.551E+07
Pu-241	1.275E+09
Sb-125	5.990E+06
Sr-90	2.500E+06
Tc-99	2.313E+07

Ft. Calhoun Fill Excavation Check Calculation

The SSDM excavation scenario discussed above assumes that the source term is excavated concrete. The fill excavation check performed by Fort Calhoun assumes that 100% of the residual radioactivity in the concrete is instantly released and uniformly mixed with the fill during basement excavation. Therefore, the source term is in the fill and not in the concrete.

The concentrations in the fill are calculated assuming that the mixing volume is 1 m^3 based on a typical fill excavation process which entails using a 1.0 to 1.5 cubic yard bucket. The activity in a 1 m^2 area of concrete is released and mixed in a single 1 m^3 bucket load (1 m distance from the wall, i.e., SA/V ratio of 1). A SA/V ratio of 1 is the same mixing assumption used in the SSDM in situ scenario conceptual model.

The objective of the check calculation was to ensure that the concentration in excavated fill after 100% of the activity on the walls is instantly released to the fill is less than the concentration in excavated concrete assuming no release. The results of this check calculation is that the concentration in the fill is less than the concentration in concrete and therefore assuming that concrete is the source term in the SSDM excavation scenario is conservative.

Ft. Calhoun SSDM Wall/Floor Initial Suite DCGL

The SSDM wall/floor DCGL is the summation of the dose from the in situ scenario, drilling spoils scenario, and excavation scenario. The same in situ and drilling spoils DCGLs apply to all basements. There is a different set of excavation scenario DCGLs for the Containment Building and all other basements due to the presence of the steel liner in Containment. Therefore, the SSDM wall/floor DCGLs for the Containment Building differ from the other basements (Auxiliary Building, Turbine Building, Circulating Water Tunnels, and Intake Structure). The SSDM wall/floor DCGLs are calculated in Excel using Equation F-7***.*** The resulting SSDM wall/floor DCGLs, not corrected for IC dose are listed in Table F-19.

$$DCGL_{wf} = \frac{1}{(\frac{1}{DCGL_i} + \frac{1}{DCGL_{ds}} + \frac{1}{DCGL_e})}$$ **Equation F-7**

where:

$DCGL_{wf}$ = SSDM wall/floor DCGL
$DCGL_i$ = SSDM in situ scenario DCGL
$DCGL_{ds}$ = SSDM drilling spoils scenario DCGL
$DCGL_e$ = SSDM excavation scenario DCGL

Table F-19: Ft. Calhoun SSDM Wall/Floor Initial Suite DCGLs (No IC Dose Correction)

Radio-nuclide	SSDM Wall/Floor DCGL ($DCGL_{wf}$) (Auxiliary, Turbine, Circulating Water Tunnels, Intake Structure) (pCi/m^2)	SSDM Wall/Floor DCGL ($DCGL_{wf}$) (Containment) (pCi/m^2)
Am-241	4.168E+06	4.046E+06
C-14	8.629E+06	8.559E+06
Ce-144	2.553E+08	5.922E+07
Cm-243	1.003E+07	7.082E+06
Cm-244	1.369E+07	1.330E+07
Co-58	3.916E+07	8.058E+06
Co-60	3.578E+06	8.640E+05
Cs-134	4.396E+06	1.345E+06
Cs-137	7.265E+06	2.749E+06
Eu-152	1.013E+07	1.983E+06
Eu-154	9.336E+06	1.839E+06
Eu-155	3.619E+08	6.791E+07
Fe-55	1.384E+10	6.308E+09
H-3	2.208E+08	2.254E+08
Ni-59	1.219E+09	1.122E+09
Ni-63	4.451E+08	4.098E+08
Np-237	7.808E+04	7.831E+04
Pu-238	2.976E+06	2.939E+06
Pu-239	2.681E+06	2.647E+06
Pu-240	2.681E+06	2.647E+06
Pu-241	1.310E+08	1.298E+08
Sb-125	2.475E+07	5.694E+06
Sr-90	8.666E+05	8.903E+05
Tc-99	5.900E+06	6.084E+06

The dose percentage attributable to the in situ, drilling spoils, and excavation scenarios from the SSDM wall/floor media can be calculated using Equation F-8.

Using Cs-137 as an example, the dose percentages attributable to the three scenarios for basements other than the Containment Building are 29%, 1%, and 70%, respectively.

$$DF_{s_j,i} = \frac{DCGL_{wf,i}}{DCGL_{s_j,i}}$$ **Equation F-8**

where:

$DF_{sj,i}$ = fraction of total SSDM wall/floor dose attributable to scenario j for radionuclide i
$DCGL_{wf,i}$ = SSDM wall/floor DCGL for radionuclide i (**Error! Reference source not found.**F-19)

$DCGL_{sj,i}$ = DCGL for wall/floor scenario j and radionuclide i

More information on the dose modeling for Fort Calhoun is given in Revision 1 of the Fort Calhoun LTP (Reference F-14).

Fort Calhoun Subsurface Structure DCGLs

At Fort Calhoun, the material to be used as basement backfill is the soil that was excavated as part of the rail spur expansion project. Approximately 132,000 cubic yards of spoils produced from this excavation are planned to be used on-site as fill material in basement structures after FSS of the structure surfaces (and embedded pipe). Because the fill material was excavated from an impacted area of the FCS site, albeit Class 3, the dose from residual radioactivity in the fill must be assessed.

The basement will be filled with the soil up to site grade (1004 feet AMSL). The backfill material will therefore have two components: the "cover" soil which is in the vadose zone above the basement structure and the "fill" soil which is within the basement structure and assumed to be in the saturated zone per the SSDM conceptual model. The dose from the "cover" soil is accounted for by the dose calculated for the Class 3 open land area represented by the fill. The dose from the "fill" soil within the basements is assumed to be fully submerged in the saturated zone. The "cover" soil is assumed to be 0.92 m (3 ft) deep and the fill soil 4 m deep consistent with the SSDM conceptual model.

The dose from the fill material within the basement is calculated as a stand-alone media, i.e., no other residual radioactivity is assumed to be within the basements. The fill material is therefore a stand-alone dose component in the compliance dose calculation**.** The fill material is assumed to contain uniform concentrations of residual radioactivity as determined through the FSS of the Class 3 open land area represented by the rail spur excavation area. Therefore, the concentrations of residual radioactivity in the "cover" component of the fill, i.e., at site grade (1004-foot elevation), is equivalent to the concentration of residual radioactivity at depth within the basements. Therefore, if the fill were excavated from the basements, the concentration of residual radioactivity in the excavated material would be equivalent to that contained at site grade. Because the dose from the soil/fill at site grade is already accounted for by the dose assessment of the open land survey unit represented by the fill no additional consideration of fill excavation, or drilling spoils, from the basement is required. However, the dose from the fully submerged fill material within the basement is a separate environmental pathway that requires assessment.

The SSDM conceptual model assumes that all of the residual radioactivity is released from the walls/floors and is then contained in the fill material. The geometry of the fill source term after release from the walls/floors in the SSDM model is equivalent to the geometry of the fill source term after placement in the basement. Therefore, the SSDM in situ scenario conceptual and mathematical models described above apply to the fill material with some minor exceptions described in the Fort Calhoun LTP.

The fill DCGL calculation assumes that all of the soil originating from the rail spur area contains uniform residual radioactivity at all depths. This is a conservative assumption for a Class 3 area where the residual radioactivity is expected to be in the 0.15 m surface layer at or near background concentrations.

The resulting DCGLs for the initial suite radionuclides and the IC adjusted DCGLs for the ROC are provided in Table F-20 and Table F-21.

Table F-20: Ft. Calhoun Fill in situ Scenario DCGLs (no IC Dose Correction)

Radionuclide	Fill in situ DCGL mrem/yr per pCi/g	Radionuclide	Fill in situ DCGL mrem/yr per pCi/g
Am-241	3.115E+00	Fe-55	1.716E+04
C-14	1.433E+01	H-3	1.733E+02
Ce-144	1.023E+03	Ni-59	1.190E+03
Cm-243	8.928E+00	Ni-63	4.345E+02
Cm-244	1.117E+01	Np-237	5.655E-02
Co-58	4.575E+02	Pu-238	2.131E+00
Co-60	1.767E+01	Pu-239	1.919E+00
Cs-134	1.197E+01	Pu-240	1.919E+00
Cs-137	1.508E+01	Pu-241	9.916E+01
Eu-152	6.108E+02	Sb-125	8.165E+01
Eu-154	4.200E+02	Sr-90	8.859E-01
Eu-155	2.703E+03	Tc-99	5.399E+00

Table F-21: Ft. Calhoun Fill in situ Scenario ROC DCGLs (IC Dose Corrected)

Radionuclide	Fill in situ DCGL Adjusted for IC Dose mrem/yr per pCi/g
C-14	1.29E+01
Co-60	1.59E+01
Cs-137	1.36E+01
Eu-152	5.50E+02

F.7 Dose Modeling for Buried Piping

As discussed in Chapter 6, 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.7.1 Dose Modeling for Buried Piping – Decommissioning Projects – 1990s and Early 2000s

Connecticut Yankee Experience

The radioactivity associated with 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 the 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 grout with the contamination from the interior surface of the piping.

Consistent with these simplified assumptions, the Derived Concentration Guidance Levels (DCGLs) calculated for the Concrete Debris Scenario (shown above) were applied to determine the dose from the slug of grout. In order to calculate the release limits for the piping [corresponding to 1 mrem/yr], 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.

Rancho Seco Experience

The buried piping scenario used by Rancho Seco incorporates the soil DCGL values discussed earlier in this Appendix. 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 was to 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.

An alternate scenario that could have been analyzed is that the contamination present on the buried piping would leach off and migrate to groundwater. This scenario could have been analyzed to see if it was more restrictive than the scenario discussed above for buried piping.

It should be noted that NRC recent guidance contained in NUREG 1757, Rev 2, states that scenarios that occur after the license is terminated at a site need to be considered. For the Rancho Seco scenarios discussed above, a possible future scenario is that the buried piping would be dug up resulting in internal and external exposures to future users of the site or persons involved in recycling the piping.

Considering the revised NRC guidance, these future scenarios would need to be analyzed to determine if lower DCGLs result than those used by Rancho Seco for buried piping.

F.7.2 Dose Modeling for Buried Pipe - Recent Experience – Ft. Calhoun

Buried piping at Fort Calhoun is defined as below ground pipe located outside of structures and basements. Note that buried electrical or structural commodities are not categorized as buried pipe and are included in the FSS of the open land survey unit in which they reside (Reference F-14).

The buried pipe to remain at license termination at Fort Calhoun is comprised predominantly of storm drains. There is also one service water piping system to remain that currently serves the maintenance shop. The total length and interior surface area the End State buried pipe are listed in Table F-22. The internal diameters of the storm drains range from 8 inches to 84 inches. The internal diameter of the service water pipe is 2.9 inches.

Table F-22: Total Length and Surface Area of Buried Piping

Piping	Length (m)	Interior Surface Area (m^2)
Storm Drain	955.9	2,167.8
Service Water	54.9	12.7
Total	1,010.8	2,181

The buried pipe was not surveyed during characterization. The pipe was to be surveyed later as a part of “continuing characterization”. However, these systems are not expected to contain residual radioactivity beyond perhaps the occasional presence of low concentrations near the detection limits.

Ft. Calhoun Buried Pipe Conceptual Model, Source Term Abstraction and Release

The buried pipe conceptual model includes two exposure scenarios:

1. the buried pipe remains in the as-left geometry (in situ scenario), and

2. the buried pipe is excavated and placed on the ground surface (excavation scenario).

The doses from the two scenarios are summed to calculate the buried pipe DCGL. The conceptual model conservatively assumes that all of the pipe is present in both the in situ and excavation scenarios simultaneously. The environmental and exposure pathways applicable to buried pipe are listed in Table F-5.

The conceptual model for the in-situ scenario assumes that the residual radioactivity on the internal surfaces of the pipe is instantly released and mixed into a 0.0254 m layer of soil over a contiguous area equal to the total internal surface area of the buried pipe (2,181 m^2). No credit is taken for the presence of the pipe to reduce environmental transport. A thin 0.0254 m mixing layer is justified because there are no mechanical mixing mechanisms when the pipes are left buried and undisturbed. The pipes are assumed to be located in the soil immediately below the 1.1 m thick vadose zone and 100% submerged in groundwater. The pipe thicknesses are ignored. The source term is therefore a 0.0254 m layer of soil in the saturated zone with a 1.1 m cover.

The conceptual model for the excavation scenario assumes that the pipes are excavated and brought to the ground surface. The residual radioactivity on the internal surfaces of the pipe is instantly released and mixed into a layer of soil on the ground surface over a contiguous area equal to the total internal surface area of the buried pipe (2,181 m^2). A sensitivity analysis is performed using 0.15 m and 1 m mixing depths. Either depth is reasonable given the extensive disturbance of the ground surface during the large-scale excavation required to remove the pipe. Assuming a vadose zone thickness of 1.1 from the site general conceptual model, the unsaturated zone thickness for the 0.15 m and 1 m mixing depths are 0.95 m and 0.1 m, respectively.

A mixing thickness of 0.15 m results in the highest dose for all initial suite radionuclides except C-14, H-3 and Tc-99. A mixing thickness of 1.0 m is therefore used for C-14, H-3 and Tc-99 and a thickness of 0.15 m for all other radionuclides.

Ft. Calhoun Buried Pipe Mathematical Model

The units of the buried pipe DCGLs are dpm/100 cm^2 to match the output of the FSS instrumentation use for buried pipe surveys. The first step in the buried pipe DCGL calculation is to calculate the concentration in soil after the release of a unit amount of activity from the pipe (1 dpm). The unitized source terms (pCi/g per dpm/cm^2) for the in situ and excavation scenarios are calculated in Excel using Equation F-9.

$$C_{s,u,i} = \frac{A_{bp,u}/2.22}{\left(SA_{Equation}t_{m,s}r_s\right)}$$ **Equation F-9**

where:

$C_{s,u.i}$ = unitized soil concentration for buried pipe scenario i (pCi/g per dpm/cm^2)
$A_{bp,u}$ = unit activity in pipe over a 1 cm^2 area (1 dpm)
2.22 = conversion factor (dpm/pCi)
$SA_{bp,u}$ = unit surface area of buried pipe (1 cm^2)
$t_{m,i}$ = thickness of soil mixing zone for buried pipe scenario *i* (in situ scenario 2.54 cm or excavation scenario 15 cm and 100 cm)
r_s = density of soil (g/cm^3)

RESRAD was used to calculate DSRs (mrem/yr per pCi/g) for both the in situ and excavation scenarios.

Ft. Calhoun Buried Pipe Excavation Scenario

The RESRAD deterministic parameters used to calculate soil DCGLs were used to calculate buried pipe excavation scenario DSRs with three changes as listed in Table F-23. The area of primary contamination is changed to 2,181 m^2 and the depth of contamination to 0.15 m or 1 m. The length parallel to aquifer flow is also changed to 26 m to reflect the 2181 m^2 area of primary contamination. The lowest DCGL, from either the 0.15 m or 1.0 m mixing distance was selected.

Table F-23: Parameter Changes to RESRAD Soil DCGL Parameter Set for Buried Pipe Excavation Scenario DSR Calculation

Parameter	Buried Pipe Excavation
Area of Contaminated Zone (m^2)	2,181
Thickness of Contaminated zone[1] (m)	0.15 m or 1 m
Length Parallel to Aquifer Flow (m)	26
Thickness of unsaturated zone[2]	0.1 m for 1 m mix thickness 0.95 m for 0.15 m mix thickness

1) Thickness of contaminated zone is the mixing depth

2) The site conceptual model vadose zone thickness is 1.1 m. The unsaturated zone thickness is 1.1 m minus the mixing depth.

Ft. Calhoun Buried Pipe In-situ Scenario

The conceptual model for the buried pipe in situ scenario is substantially different from the soil conceptual model. A 1.1 m cover is present and the source term is a thin layer that is fully submerged. Therefore, unlike the buried pipe excavation scenario, a separate uncertainty analysis was performed for the in-situ scenario.

The process for performing the uncertainty analysis is the same as that used for soil. The soil deterministic parameters and PDFs were used with modifications as shown in Table F-24. The deterministic parameters in Table F-24 (except for the primary radionuclide K_d values) are also used in the RESRAD DSR calculations.

Table F-24: Changes to Soil Uncertainty Analysis Parameter Set in Attachment 6-Required for Buried Pipe in situ Scenario Uncertainty Analysis

Parameter	Buried Pipe in situ Scenario
Cover depth (m)	1.1
Cover erosion rate (m/y)	RESRAD Default PDF
Contaminated zone erosion rate[1]	0
Area of contaminated Zone (m^2)	2,181
Thickness of contaminated zone (m)	0.0254 m
Length parallel to aquifer flow (m)	26
Contaminated fraction below the water table	1
Contaminated zone density (g/cm^3)	1.49
Contaminated zone total porosity (unitless)	0.45
Contaminated zone field capacity (unitless)	0.24
Contaminated zone hydraulic conductivity (m/yr)	4350
Contaminated zone deterministic K_d values changed to the 50th percentile of PDF in ***Omaha Public Power District, "FC-20-012, Fort Calhoun Station Decommissioning Project Radiological Characterization Report***	Radionuclide Dependent
Contaminated zone K_d PDFs changed to the distributions in Table 2.13.1 or 2.13.10 of ***Omaha Public Power District, "FC-20-012, Fort Calhoun Station Decommissioning Project Radiological Characterization Report***	Radionuclide Dependent PDF

1) Contaminated zone is completely submerged in saturated zone

The changes to the contaminated zone hydrogeological parameters and K_d values are required to satisfy the conceptual model which assumes that the buried pipe is fully submerged in the saturated zone as opposed to being in the vadose zone.

Table F-25 lists the parameters that are stochastic and require assessment by uncertainty analysis for the buried pipe in situ scenario. The uncertainty analysis results and selected deterministic parameters to be applied in the RESRAD DSR calculation are also listed in Table F-25.

Table F-25: Buried Pipe in situ Scenario RESRAD Uncertainty Analysis Results for Non-Nuclide Specific Parameters and Selected Deterministic Values

Parameter	Correlation to Dose	Radionuclide	Selected Deterministic Value
Cover erosion rate (m/yr)	positive	Am-241, Co-60, Cs-134, Cs-137, Eu-152, Eu-154, Eu-155, Pu-238, Pu-239, Pu-240, Pu-241, Sb-125	2.92E-03
Contaminated zone b parameter (unitless)	NS[1]	NA	3.6
Evapotranspiration coefficient (unitless)	positive	C-14[2], Pu-241[2], H-3, Tc-99	0.87
Wind Speed (m/s)			3.75
Runoff coefficient (unitless)	positive	H-3[2], Tc-99	0.63
Depth of roots (m)	negative	Am-241, Ce-144, Cm-244, Pu-239, Pu-241, Pu-241, Fe-55[2], Ni-59,	1.23
Well pump intake depth (m)	negative	Am-241, C-14, Cm-243[2], Cm-244, Pu-238, Pu-239, Pu-240, Pu-241, Fe-55, H-3, Np-237, Ni-59[2], Tc-99	21.4
Mass loading for inhalation (g/m^3)	NS	NA	2.35E-05
Indoor dust filtration factor (unitless)	NS	NA	0.55
Depth of Soil Mixing Layer (m)	NS	NA	0.23
Wet foliar interception fraction of leafy vegetables (unitless)	NS	NA	0.58
Weathering removal constant all vegetation (unitless)	NS	NA	33
Wet weight crop yield of fruit, grain, and non-leafy vegetable (kg/m^2)	NS	NA	1.75

Table F-26: Buried Pipe in situ Scenario RESRAD K_d Parameter Uncertainty Analysis Results for Contaminated Zone and Saturated Zone and Selected Deterministic Values[1]

Radio-nuclide	Correlation to Dose	Basis of Deterministic Parameter Selection	Selected Deterministic Value (cm^3/g)
Am-241	negative	25th percentile	269
C-14	negative	25th percentile	1.26
Ce-144	NS[2]	50th percentile	399
Cm-243	negative	25th percentile	572
Cm-244	negative	25th percentile	572
Co-58	NS	50th percentile	260
Co-60	NS	50th percentile	260
Cs-134	NS	50th percentile	528
Cs-137	NS	50th percentile	528
Eu-152	NS	50th percentile	829
Eu-154	NS	50th percentile	829
Eu-155	NS	50th percentile	829
Fe-55	NS	50th percentile	321
H-3	negative	25th percentile	0.043
Ni-59	NS	50th percentile	130
Ni-63	NS	50th percentile	130
Np-237	negative	25th percentile	5.49
Pu-238	negative	25th percentile	156
Pu-239	negative	25th percentile	156
Pu-240	negative	25th percentile	156
Pu-241	negative	25th percentile	156
Sb-125	NS	50th percentile	16.9
Sr-90	NS	50th percentile	22
Tc-99	negative	25th percentile	0.019

1) The buried pipe is fully submerged in the saturated zone. An unsaturated zone is not present.

2) not sensitive (NS)

The deterministic parameters listed in Table F-24 and those selected through uncertainty analysis and listed in Table F-25 and Table F-26, are used in RESRAD to calculate the buried pipe in situ scenario DSRs. Before the RESRAD runs were made, two additional sensitivity analyses were performed using the DSR parameter set.

First, the performance of the RESRAD non-dispersion model for the buried pipe in situ scenario, which assumes that the source term is fully submerged, was checked to ensure the results are applicable and reasonable. The results of this calculation described fully in the Fort Calhoun LTP validate the use of the RESRAD non-dispersion groundwater model for the fully submerged buried pipe in situ scenario source term.

A second sensitivity analysis was performed to ensure that placing the source in the saturated zone is conservative. At Fort Calhoun, the water table elevation varies seasonally with the stage of the Missouri river. Most of the year the water table elevation is expected to be lower than 1.1 m bgs. The buried pipe could therefore be in the vadose zone for most of the year. A sensitivity analysis was performed by placing the 0.0254 m thick source into the vadose zone (i.e., setting the contaminated zone fraction below the water table to 0). The thickness of the unsaturated zone was set to a thin layer of 0.1 m. Using Cs-137 as Sr-90 as examples, the DSRs with a fully submerged source term are slightly higher (less than 2%). The fully submerged source term is conservative albeit to a minor extent.

Ft. Calhoun Buried Pipe Initial Suite DCGL

DCGLs for the buried pipe excavation scenario are calculated for both a 0.15 and 1.0 m mixing distance. The DSR increases with increasing thickness of the contaminated zone (mixing thickness) when the source term concentration is constant but the concentration decreases as10 a function of mixing thickness. The lowest DCGL, from either the 0.15 m or 1.0 m mixing distance was selected. For the in situ scenario, the DCGL is based on a thickness of 0.0254 m. The DCGLs for both the in situ and excavation scenarios are calculated in Excel using Equation F-10.

$$DCGL_{bp,s,i} = \left(\frac{25}{C_{bp,u,s}DSR_{bp,i}}\right)100$$ **Equation F-10**

where:

$DCGL_{bp,s,i}$ = buried pipe DCGL for scenario *s* and radionuclide *i* (dpm/100 cm^2)
25 = 25 mrem/yr dose criterion
$C_{bp,u,s}$ = unitized soil concentration for buried pipe scenario *is* calculated using **Error! Reference source not found.**F-9 (pCi/g per dpm/cm^2)
$DSR_{bp,i}$ = buried pipe DSR for radionuclide i (mrem/yr per pCi/g)
100 = 100 cm^2 to calculate the DCGL in units of dpm/100 cm^2.

For the excavation scenario, a mixing thickness of 0.15 m results in the highest dose for all initial suite radionuclides except C-14, H-3, and Tc-99. A mixing thickness of 1.0 m is therefore used to calculate excavation DCGLs for C-14, H-3, and Tc-99 and a thickness of 0.15 m is used for all other radionuclides. The buried pipe in situ and excavation DCGLs (with no IC dose correction) are listed in **Error! Reference source not found.**

Table F-27: Buried Pipe Initial Suite Excavation and in situ Scenario DCGLs (No IC Dose Correction)

Radionuclide	Buried Pipe Excavation Scenario DCGL (dpm/100 cm^2)	Buried Pipe In situ Scenario DCGL (dpm/100 cm^2)
Am-241	7.442E+05	7.905E+05
C-14	3.596E+06	8.874E+06
Ce-144	1.430E+06	1.213E+08
Cm-243	3.511E+05	1.387E+06
Cm-244	1.548E+06	1.736E+06
Co-58	1.898E+05	4.746E+07
Co-60	2.103E+04	1.836E+06
Cs-134	3.670E+04	1.238E+06
Cs-137	8.432E+04	1.558E+06
Eu-152	4.610E+04	2.653E+08
Eu-154	4.282E+04	1.826E+08
Eu-155	1.611E+06	1.185E+09
Fe-55	1.341E+09	5.346E+09

Radionuclide	Buried Pipe Excavation Scenario DCGL (dpm/100 cm^2)	Buried Pipe In situ Scenario DCGL (dpm/100 cm^2)
H-3	5.409E+07	6.443E+07
Ni-59	2.400E+08	2.817E+08
Ni-63	8.763E+07	1.029E+08
Np-237	3.955E+04	1.987E+04
Pu-238	9.576E+05	7.360E+05
Pu-239	8.624E+05	6.626E+05
Pu-240	8.630E+05	6.626E+05
Pu-241	3.018E+07	2.766E+07
Sb-125	1.382E+05	2.024E+07
Sr-90	8.093E+04	8.279E+04
Tc-99	5.938E+05	1.475E+06

The excavation scenario and in situ scenario DCGLs are summed to calculate the final buried pipe DCGL in Excel using Equation F-11. The resulting DCGLs are listed in Table F-28. The DCGLs in Table F-28 are not corrected for IC dose.

$$DCGL_{bp,i} = \frac{1}{({}^{1}/_{DCGL_{bpi,i}} + {}^{1}/_{DCGL_{bpe,i}})}$$ **Equation F-11**

where:

$DCGL_{bp,i}$ = Buried pipe DCGL for radionuclide i
$DCGL_{bpi,i}$ = Buried pipe in situ scenario DCGL for radionuclide i
$DCGL_{bpe,i}$ = Buried pipe excavation scenario DCGL for radionuclide i

Table F-28: Buried Pipe Initial Suite DCGLs (No IC Dose Correction)

Radionuclide	$DCGL_{bp}$ (dpm/100 cm^2)	Radionuclide	$DCGL_{bp}$ (dpm/100 cm^2)
Am-241	3.833E+05	Fe-55	1.072E+09
C-14	2.559E+06	H-3	2.941E+07
Ce-144	1.413E+06	Ni-59	1.296E+08
Cm-243	2.802E+05	Ni-63	4.732E+07
Cm-244	8.182E+05	Np-237	1.323E+04
Co-58	1.890E+05	Pu-238	4.161E+05
Co-60	2.079E+04	Pu-239	3.747E+05
Cs-134	3.564E+04	Pu-240	3.748E+05
Cs-137	7.999E+04	Pu-241	1.443E+07
Eu-152	4.609E+04	Sb-125	1.372E+05
Eu-154	4.281E+04	Sr-90	4.093E+04
Eu-155	1.609E+06	Tc-99	4.234E+05

F.8 Dose Modeling for Embedded Piping

F.8.1 Dose Modeling for Embedded Piping – Decommissioning Projects – 1990s to Early 2000s

Trojan Plant Experience

EPRI Report "Remediation of Embedded Piping, Trojan Nuclear Plant Decommissioning Experience," (EPRI Report # 1000908 Reference F-13), provides a detailed discussion of experiences with embedded piping at the Trojan Plant. The following are the highlights of that report unless other references are provided.

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 were 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.

Trojan chose to clean and/or survey in place 13,700 feet (4,179 meters) of the embedded piping to meet final site survey acceptance criteria (Reference F-16). Specialized survey instrumentation allowed the site to perform in situ surveys on the piping to verify it to be acceptably clean. Trojan demonstrated various decontamination and primarily used grit blasting to decontaminate the piping where needed.

Trojan also demonstrated a number of different survey techniques, mostly using small Geiger-Mueller detectors (GM) and Gas Flow Proportional Detectors (GFPD). Most of the surveys were performed with an array of GM detectors, some arrays as long as 100 feet with ability go around 45- or 90-degree bends.

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. Reference F-13 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.

Trojan Dose Modeling Approach for Embedded Piping

Portland General Electric used two approaches to determine the release limits for embedded piping at Trojan.

For the Turbine Building, the NRC Screening Values (developed using the Building Occupancy Scenario) were used to calculate gross activity DCGLs. After screening out radionuclides whose MDC was lower than 10% of their DCGLS and one outlier result for Ni-63 in characterization samples, the only radionuclides used in the DCGL calculation were Cs-137 at 88.6 % and Co-60 at 11.4 %. Applying the fractions of the NRC Screening values, a weighted average gross activity DCGL of 21,000 dpm/100 cm^2 was calculated for the Trojan Turbine Building embedded piping (Reference F-16).

Per Reference F-13 for the Auxiliary/Fuel Building, the embedded piping was:

First remediated using grit blasting to remove loose surface contamination. This would eliminate inhalation of airborne contamination as a dose pathway. Then filled with a grout/concrete mix to immobilize remaining fixed radioactivity

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:

Per Reference F-17, when contaminated embedded piping is present in a building survey unit, the building surface residual radioactivity was limited to the non-embedded dose fraction of the screening DCGL value. For Auxiliary/Fuel Building, this is the equivalent of 20 mrem/yr, which allows the dose from the embedded pipe exposure pathway to contribute up to 5 mrem/yr. Based on shielding calculations and the assumption that the uniform residual surface activity on the internal surfaces of embedded pipes is less than or equal to 100,000 dpm/100 cm^2 beta-gamma, the dose contribution from embedded piping is less than 5 mrem/yr. Therefore, if the average measured residual radioactivity from TNP building surfaces results in a dose of greater than or equal to 20 mrem/yr or the average measured residual radioactivity inside embedded pipes results in a dose of greater than or equal to 5 mrem/yr, an evaluation is performed to determine if the site release dose criterion would be exceeded.

Although this methodology was acceptable for use at Trojan at the time, more recent NRC guidance recommends that alternate scenarios such as removal of the piping for recycle be evaluated to determine if the dose from that scenario, which would include the dose from inhalation, is higher than the direct exposure pathway included in the scenario discussed above.

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.

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

It should be noted that some of the experiences summarized above assume only dose from direct exposure from gamma radioactivity. When dose modeling for embedded piping, exposures from other types of contamination (in particular alpha radioactivity) and via other pathways (in particular inhalation) need to be evaluated.

F.8.2 Dose Modeling for Embedded Pipe - Recent Experiences

Fort Calhoun SSDM Embedded Pipe Source Term, Conceptual Model, and Mathematical Model

The Auxiliary Building basement and Turbine Building basement at Fort Calhoun contain floor drains that are embedded in concrete (embedded pipe). Embedded pipe at Fort Calhoun is defined as pipe that runs vertically through a concrete wall or horizontally through a concrete floor. The floor drains in the Turbine Building basement do not meet the standard definition of embedded pipe. The drains are open to the basement but run under the basement slab. The conceptual model for Auxiliary Building and Turbine Building drains is described below (Reference F-14).

The survey of embedded pipe was not included in the characterization program due to access issues. The Auxiliary Building floor drains are known to contain elevated levels of contamination. The Turbine drains are expected to contain minimal contamination. A radiological assessment survey of Auxiliary Building drains will be performed after the initial pass of decontamination by high pressure hydrolazing. The characterization of the Turbine Building drains was to be performed as a part of the continuing characterization program. Decontamination is not expected to be required for the Turbine Building drains.

The conceptual and mathematical models for the embedded pipe dose assessment are fundamentally the same as the SSDM wall/floor in situ model described above. The activity on the internal surface of the pipes is assumed to instantly release and mix into a 1 m thick layer of fill above the floor.

There are three floor elevations that contain embedded pipe: the 989 and 971 foot elevations of the Auxiliary basement and the 990 foot elevation of the Turbine basement. These three floor elevations are treated as separate areas for the embedded pipe DCGL calculations because the DCGL is dependent on the ratio of pipe internal surface area to the floor area as shown in Equation F-12. Details on these dimensions are contained in the Fort Calhoun LTP.

The DCGL is calculated in three steps:

1. calculate the concentration in fill after release from the pipe (pCi/g per pCi/m^2) using Equation F-12,

2. calculate DSRs with RESRAD using the deterministic parameters developed for the SSDM in situ scenario with the changes summarized below, and

3. calculate the embedded pipe DCGLs using Equation F-13.

$$C_{f,e_i} = \frac{A_{ep,u} SA_{ep,e_i} 0.0929}{SA_{f,e_i} 0.0929 D_m 1x10^6 \rho_f}$$ **Equation F-12**

where:

$C_{f,ei}$ = concentration in fill from release of activity from embedded pipe at floor elevation i (pCi/g per pCi/m^2)
$A_{ep,u}$ = unit activity in embedded pipe (1 pCi per m^2),
$SA_{ep,ei}$ = embedded pipe internal surface area in floor elevation i (ft^2),
0.0929 = Conversion Factor (m^2/ft^2)
$SA_{f,ei}$ = floor surface area at elevation i (ft^2),
D_m = mix distance in fill (1 m)
$1x10^6$ = Conversion factor (g/cm^3)
r_f = bulk density of fill (assumed to be sand) (g/cm^3)

The embedded pipe DSRs are calculated using the deterministic parameters developed for the SSDM in situ scenario with the changes discussed in the Fort Calhoun LTP. The uncertainty analysis conducted for the SSDM in situ scenario is assumed to apply to embedded pipe given that both involve a fully submerged source term in a basement under a cover. The total depth of the fill is assumed to be the same as for the SSDM in situ scenario, i.e., 4 m. The cover depth for the embedded pipe scenario is 3.92 m which is the depth of the cover over the backfilled basements (0.92 m) plus the 3 m of fill assumed to be above the 1 m floor mixing zone.

The conceptual model applies directly to the Auxiliary Building basement embedded pipe which runs within the concrete foundation. However, the Turbine Building basement floor drains run below the slab and have the attributes of buried pipe as well as embedded pipe. Treating the Turbine Building floor drains as buried pipe leads to the release of the activity from the internal surface of the pipe to the surrounding soil under the slab. Treating the Turbine Building floor drains as embedded pipe leads to release of activity from the pipe through openings in the floor into the basement.

The buried pipe conceptual model assumes a mix distance of 0.0254 m and a well depth of 21.4 m. Both models assume the source term is fully submerged in the saturated zone. Therefore, the differences between the buried pipe and embedded pipe conceptual models are essentially reduced to:

1. the mixing distance into the soil or fill after release from the pipe (0.0254 m for buried pipe and 1 m for embedded pipe) and

2. the assumed depth of the well (21.4 m for buried pipe and 4 m for embedded pipe).

A check calculation was conducted using the RESRAD parameter set developed for embedded pipe with the buried pipe mixing depth and well depth of 0.0254 m and 21.4 m, respectively. The DSRs with the embedded pipe mixing depth and well depth are higher than the DSRs with the buried pipe mixing and well depths for all radionuclides and are therefore applied in the embedded pipe DSR calculation for the Turbine basement embedded pipe.

The embedded pipe DCGLs are calculated for each floor elevation using Equation F-13. The resulting embedded pipe DSRs and DCGLs are listed in Table F-29**.**

$$DCGL_{ep,e_i,j} = \frac{25}{DSR_{ep,j}C_{f,e_i}}$$ **Equation F-13**

Where:

$DCGL_{ep,ei,j}$ = embedded pipe DCGL at floor elevation i for radionuclide j (pCi/m^2)
25 = 25 mrem/yr dose criterion
$DSR_{ep,j}$ = embedded pipe DSR for radionuclide j (mrem/yr per pCi/g)
$C_{f,ei}$ = concentration in fill from release of activity from embedded pipe at floor elevation i (Equation F-12) (pCi/g per pCi/m^2)

Fort Calhoun Embedded Pipe Initial Suite DCGL

The DSRs and the results of the embedded pipe DCGL calculations (not corrected for IC dose) are provided in Table F-29.

Table F-29: Embedded Pipe Initial Suite DSRs and DCGL (No IC Dose Correction)

Radionuclide	DSR (mrem/yr per pCi/g)	$DCGL_{ep}$ 971¢ Auxiliary Floor Drains (pCi/m^2)	$DCGL_{ep}$ 989¢ Auxiliary Floor Drains (pCi/m^2)	$DCGL_{ep}$ 990¢ Turbine Floor Drains (pCi/m^2)
Am-241	1.851E+00	4.900E+08	3.83E+08	2.63E+08
C-14	5.787E-01	1.568E+09	1.224E+09	8.406E+08
Ce-144	4.811E-03	1.886E+11	1.473E+11	1.011E+11
Cm-243	5.938E-01	1.528E+09	1.193E+09	8.192E+08
Cm-244	4.747E-01	1.912E+09	1.493E+09	1.025E+09
Co-58	6.528E-03	1.390E+11	1.085E+11	7.452E+10
Co-60	1.691E-01	5.366E+09	4.190E+09	2.877E+09
Cs-134	1.365E-01	6.648E+09	5.191E+09	3.564E+09
Cs-137	1.084E-01	8.371E+09	6.537E+09	4.487E+09
Eu-152	9.568E-03	9.484E+10	7.406E+10	5.084E+10
Eu-154	1.392E-02	6.519E+10	5.090E+10	3.495E+10
Eu-155	2.164E-03	4.193E+11	3.274E+11	2.248E+11
Fe-55	3.283E-04	2.764E+12	2.158E+12	1.482E+12
H-3	3.725E-02	2.436E+10	1.902E+10	1.306E+10
Ni-59	4.373E-03	2.075E+11	1.620E+11	1.112E+11
Ni-63	8.698E-03	1.043E+11	8.146E+10	5.593E+10
Np-237	1.476E+02	6.148E+06	4.801E+06	3.296E+06
Pu-238	2.796E+00	3.246E+08	2.534E+08	1.740E+08
Pu-239	4.086E+00	2.221E+08	1.734E+08	1.190E+08
Pu-240	3.782E+00	2.399E+08	1.874E+08	1.286E+08
Pu-241	6.079E-02	1.493E+10	1.166E+10	8.002E+09
Sb-125	6.833E-02	1.328E+10	1.037E+10	7.119E+09
Sr-90	4.010E+00	2.263E+08	1.767E+08	1.213E+08
Tc-99	1.151E+00	7.884E+08	6.156E+08	4.226E+08

F.9 Buried Materials

Additional NRC guidance on addressing dose modeling for buried materials is contained in NUREG 1757, Vol 2, Rev 2 Appendices I and J. Additional NRC guidance concerning buried materials is contained in Reference 5-6.

Experiences with the dose modeling for buried concrete at Zion is contained in Section F.6.3.

F.10 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 U.S. Nuclear Regulatory Commission, NUREG/CR-6692, “Probabilistic Modules for the RESRAD and RESRAD-BUILD Computer Codes,” November 2000.

F-7 U.S. Nuclear Regulatory Commission, NUREG/CR-6676, “Probabilistic Dose Analysis Using Parameter Distributions Developed for RESRAD and RESRAD-BUILD Computer,” May 2000.

F-8 EPRI Report # 1015502. “Concrete Characterization and Dose Modeling During Plant Decommissioning: Detailed Experiences 1993 – 2007,” March 2008.

F-9 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-10 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-11 Zion Station Restoration Project License Termination Plan, Chapter 6, Revision 2.

F-12 Argonne National Laboratory, ANL/EAD/03-01, “User’s Manual for RESRAD-BUILD Version 3,” June 2003.

F-13 EPRI Report # 1000908, “Remediation of Embedded Piping, Trojan Nuclear Plant Decommissioning Experience,” dated October 2000.

F-14 Fort Calhoun Station, License Termination Plan, Rev 1, December 6, 2023.

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

F-16 Portland General Electric Company Letter VPN-063-2004, Trojan Nuclear Power Plant Final Survey Report for Embedded Piping, dated November 10, 2004.

F-17 EPRI Report # 1003423, Trojan Nuclear Plant License Termination Plan Development Project, April 2002.

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.

The analysis results of the first round of groundwater sampling under the detailed characterization plan showed elevated levels of Strontium-90 (Sr-90) in some of the wells near to and hydraulically downgradient to the tank farm previously mentioned. Additionally, there were detections of H-3 consistent with previous sampling.

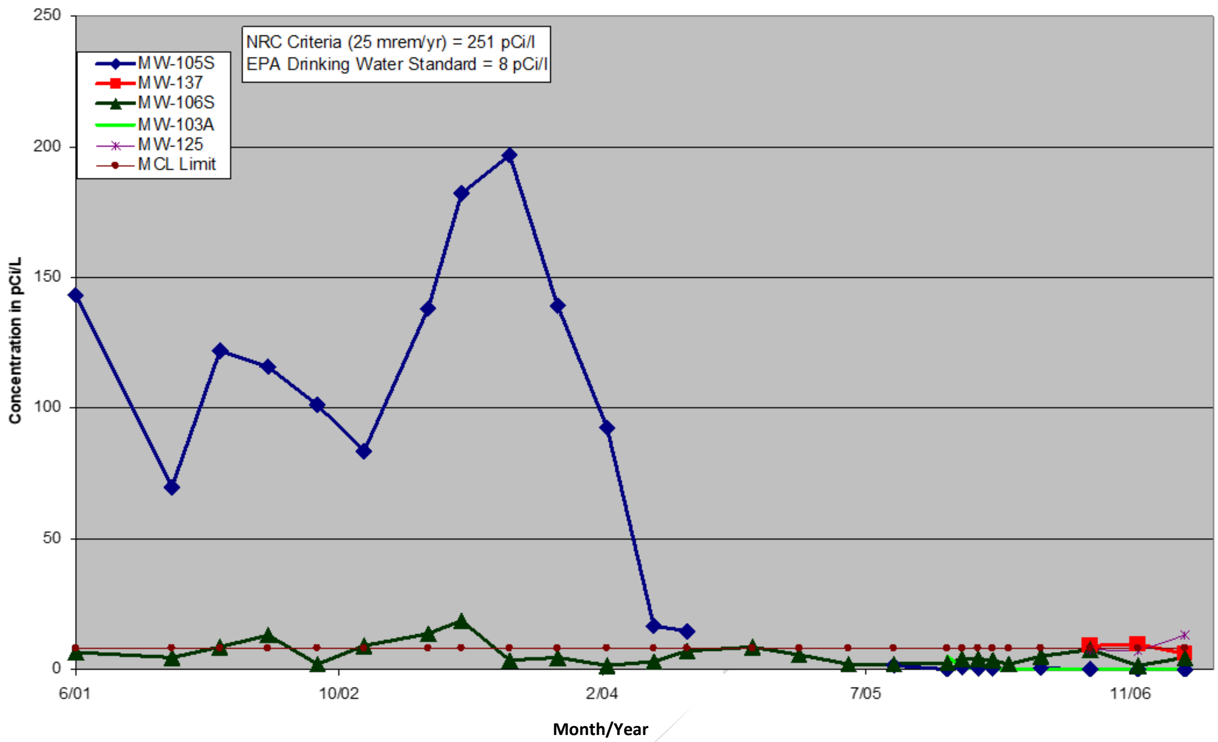


Figure G-1: Strontium-90 Concentrations in Groundwater at CY (Concentration units are pCi/L)

Unlike the rapid drop of H-3 concentrations from their historical highs, the pre-remediation trend of Sr-90 concentrations shown in Figure G-1 appears more erratic. A more detailed analysis of the Sr-90 trend showed a correlation of the increases in Sr-90 groundwater concentrations with increasing water table levels. This fact indicated that a continuing inventory of Sr-90 existed at some locations on the site and that this inventory was being contacted during periods of high-water table (generally springtime periods of snow melt and precipitation). It was postulated that as the water table rose, it encountered contaminated soil that was above the average water table level. This contact would result in the dissolution of the Sr-90 from the soil due to the chemical equilibrium of water and soil. This mobilized Sr-90 would then move through the groundwater and be detected in the samples taken from wells downgradient of the contaminated soil inventory. The early 2004 large reduction in concentration on Figure G-1 corresponds to area dewatering activities. This reduction indicated that the source of the Sr-90 contamination in groundwater had been separated from the groundwater level.

G.3 Soil Characterization

To prove this theory, an extensive soil characterization program was initiated. Approximately 200 sample locations generating approximately 1000 samples, primarily in the tank farm and downgradient areas, were chosen. Soil in other areas of the plant was also analyzed to determine if this situation existed elsewhere. Many samples' locations were outside buildings but for certain buildings the soil under the building was sampled by boring through floors. The direct push equipment was used extensively for this sampling. The results of this campaign indicated the highest levels of radionuclides were present under the tank farm. In addition to high levels of Sr-90, detectable levels of most of the other radionuclides of concern at CY were present in the soil below the tank farm. At the locations where the highest levels of Sr-90 contamination were found in groundwater, the levels of Sr-90 in the soil were significantly lower than those in the tank farm and the contamination was not present above the water table elevation. This information supported the theory of the movement of Sr-90 discussed above.

G.4 Soil Remediation

For compliance with the CY LTP, at least an 18-month groundwater monitoring period (including two high water table springtime periods) was needed after the remediation of the source of the groundwater contamination. Considering this factor, it was important to remove the source so as not to delay removal of the area from the NRC license. Remediation of soil was conducted to also meet the U.S. EPA Maximum Contaminant Levels (MCL also known as the EPR Drinking Water Standards). The structures in the tank farm area and the PAB were removed in their entirety due to:

- The presence of soil requiring removal under portions of these structures
- The concrete in these structures was contaminated to the point that decontamination to meet the DCGLs was determined to be more expensive than simply removing all the concrete.

G.5 Post Remediation Radiological Assessment

Once the remediation of the excavation was thought to be complete, a radiological assessment was performed using the following protocol:

- Remediation was conducted until the perimeter soil of the excavation met the DCGLs and the groundwater screening concentrations needed to meet the EPA MCLs.
- As most of the soil that was inside the perimeter exceeded either the DCGLs or the groundwater screening concentrations, essentially all soil was removed. Although this was performed primarily with excavators, the final removal of soil to bedrock was performed with a high flow rate vacuum truck. The radioactivity inventory for any incidental soil left in the excavation was determined by a volume estimate and soil sampling.
- Any concrete structures that formed the perimeter of the excavation or concrete footing that were left inside the excavation were sampled using coring. As with the incidental soil, a radioactivity inventory was calculated for this concrete.
- Core bore samples were taken from the bedrock in the bottom of the excavation. In the western 2/3 of the excavation, there was essentially no radioactivity in the bedrock. The analysis results for the bedrock showed very low concentrations of Sr-90. Although these were determined to be most likely false positive detections, a radioactivity inventory was calculated for the bedrock using the values for conservatism.
- Scanning of the bedrock was conducted using sodium iodine detectors, beta sensitive probes, and a portable gamma spectroscopy system. No plant related radioactivity was detected during any of the scanning techniques.

- The area of the excavation that corresponded to the Residual Heat Removal Pit portion of the excavation was an area of bedrock that had been blasted to a depth of 50 foot (15.2 m) below grade during plant construction. The vertical rock faces on the sides of the pit were up to 40 feet (12.2 m) tall. The scanning of this area could not be accomplished safely with conventional scanning techniques. The portable gamma spectroscopy system positioned by a crane was used to remotely assess the radioactivity inventory of this area. The In-Situ Gamma Spectroscopy results indicated no detectable plant related radioactivity.

The radioactivity inventory determined by the above assessment techniques was compared to the groundwater related site release criteria as follows:

- The sum of the radionuclide inventories of soil inside the excavation perimeter, concrete structures that bound or were inside the excavation, and in the first two inches (5 cm) of bedrock were conservatively assumed to be instantaneously released into the excavation.

- This released inventory was next assumed to mix with the groundwater and engineered backfill present in the excavation. The equilibrium concentrations of the radionuclides in groundwater were calculated using the distribution coefficients (K_ds) for the engineered backfill.

- The equilibrium groundwater concentrations were next compared to the LTP DCGLs to be applied at NRC license termination and the EPA MCLs per the EPA/NRC MOU.

G.6 Tank Farm Portion of Primary Auxiliary Building (PAB) Excavation

The remediation of the last (most easterly third) portion of the PAB Excavation proved to be more difficult than the first two thirds. Due to blasting during plant construction, the bedrock in this area was more fractured than in other areas of the PAB excavation. As the area was where most of the leakage events affecting groundwater had occurred, soil in this area had the highest concentrations measured on site. More radionuclides were detected in this area as the soil retarded movement of most of the radionuclides away from the leakage areas.

The same remediation techniques were used as for the remainder of the PAB excavation in this area. After the soil had been vacuumed from the area directly below the tank farm, radioactivity could be detected in bedrock fractures at levels up to 40 mrem/hr (0.4 mSv/hr). The rock in the areas of the "hot spots" was broken up with a hydraulic hammer mounted on an excavator (Hoe-ram) and disposed of as radwaste. The newly exposed fractures exhibited lower but still elevated levels up to 1 mrem/hr (0.01 mSv/hr). It was decided that rather than continuing to "chase" contamination in the fractures, a comprehensive survey of the bedrock in this area would be conducted. This survey consisted of the following tasks:

- Samples of the granular material in the fractures that contained the radioactivity were analyzed for radionuclides and leaching tests were performed to determine distribution coefficients (K_ds).

- In the area of the hot spots, bedrock cores were drilled to depths up to 25 feet (7.6 m) below the bedrock surface. These cores were scanned, and wafers cut from the locations exhibiting higher radioactivity levels. These wafers were sent to an off-site laboratory for radionuclide evaluation. Additionally, down-hole gamma logging using a sodium iodine detector was used to determine areas of higher radionuclide concentrations.

- The down-hole gamma logging work showed that area in the bedrock could be assessed without the time and expense of drilling cores and the subsequent sample analysis. The bounds of the contamination were evaluated using boreholes drilled using a rock coring machine. This machine was much more effective in drilling through the granite and pegmatite which made up most of the bedrock in this area. The resulting holes were up to 30 feet (9.14 m) deep and were assessed using the down-hole gamma technique described above.

The results of the comprehensive survey were that an area approximately 25 feet (7.6 m) square and up to 16 feet (4.9 m) deep needed to be removed to ensure that groundwater would not be substantially affected by the radioactivity contained in the granular material in the bedrock fractures. It was decided that blasting (like that used in road construction) was the most effective method to remediate the bedrock area identified. This blasting was performed, and the exposed area surveyed. The results of that survey showed the following:

- A fracture containing elevated levels of radioactivity passed under one of the footings of the yard crane. As the other end of the yard crane was still in use, a structural evaluation of the crane was performed to assess the removal of that footing. It was determined that with the installation of cable ties to other yard crane legs, the footing in question could be removed. Another adjacent footing had been removed earlier in the remediation. The supports were installed, and the footing removed.

- Granular material from blasting (granulite) contained detectable radioactive material and required removal via vacuuming.

- Additional fracture material containing elevated levels of Sr-90 was identified in an area below an area of previous remediation. This area was broken up with a hoe-ram and disposed of as radioactive material.

A post remediation scan survey was performed on this additional area of remediation and no readings over background were found. The results of a post remediation radiological assessment were that projected groundwater concentrations would be well below all the site release criteria.

APPENDIX H. DISCRETE RADIOACTIVE PARTICLES (DRP)

H.1 Introduction

Discrete radioactive particles (DRPs) are small, on the order of 1 mm, such as small chips of biological shield materials or small pieces of metal from reactor vessel/internals segmentation. With adequate isolation and controls in place, any discrete particles should typically be picked up during contamination surveys and controlled. If discrete particles are released beyond the radiologically controlled areas, the NRC is giving credit for a licensee's decontamination efforts if they have performed an adequate survey for DRPs in an area where the release has occurred. All DRPs should have been identified/removed prior to FSS. The NRC should be informed if any DRPs are collected during FSS and scanning survey methods for suspect survey units should be adjusted, to the extent practical, to be as sensitive as possible to ensure maximum sensitivity to DRPs. Areas where DRPs were previously remediated on the site (primarily outside surface areas) should have increased sensitivity monitoring during the final status survey.

DRPs are beta emitting, and although highly radioactive, produce a dose distribution that is both highly non-uniform and localized and are intrinsically different than the diffuse contamination discussed in MARSSIM and evaluated using RESRAD. The biological effects of a localized, non-uniform field on the skin are qualitatively different from the biological effects resulting from relatively uniform irradiation of large areas of the skin and these effects are considered much less severe. (U.S. NRC SECY-98-245, Rulemaking Plan – Protection Against Discrete Radioactive Particle (DRP) Exposures (10 CFR PART 20), October 23, 1998.)

The NRC recognizes that DRP generation is intrinsic to the process of decommissioning a nuclear power plant. Robust control of discrete particles at the source during decommissioning should be sufficient to resolve concerns.

H.2 Minimizing the Likelihood of DRPs during Decommissioning

The primary source of DRPs during decommissioning is from fuel failures (pre-decommissioning existence), activated metal segmentation, and activated concrete demolition/size reduction. Each of these are discussed in the following paragraphs.

H.2.1 Fuel DRP

Generally, one of the earliest activities during decommissioning is the transfer of remaining spent fuel in the fuel pool to dry cask storage. More than likely, this is a continuation of transfers that have been previously occurring during operation and no new risks are encountered. However, once all fuel has been transferred, the cleanup of the spent fuel pool under water, drain down of spent fuel pool water, and removal and packaging of spent fuel pool racks may present enhanced requirements for DRP controls.

Typically fuel DRPs are not of sufficient size to present a whole-body dose concern. However, if past activities in the spent fuel pool included work where fuel debris or activated metal fragments could be present, dose rate scans should be performed for worker dose controls both under water and during drain down.

Fuel DRP controls may be linked with alpha-emitting radiation hazard controls when it comes to spent fuel pool work. Perform as much work as possible underwater while maintaining control of the items and equipment that are being removed from the water. During underwater hydrolazing and vacuuming, maintain water clarity and maximize radioactivity reduction of the water through water filtration, both local and fuel pool cleanup if available. Maintain water cleanup as long as possible during drain down while hydrolazing walls and spent fuel racks. For extreme alpha hazard situations, consider the use of divers to coat spent fuel pool surfaces prior to drain down.

After drain down activities, spent fuel rack radiological surveys will determine the dose rates, contamination levels and appropriate control measures required for relocation, downsizing and/or packaging of the racks. Application of fixative coatings to the racks may be necessary to minimize the spread of contamination during packaging activities prior to disposal. Tenting and localized airborne controls may be utilized if significant downsizing is required.

H.2.2 Activated Metal DRP

The most radiologically activated metals are the reactor vessel internal components. Due to high dose rate concerns for workers, most of the reactor internals are segmented underwater. This helps to contain DRPs generated during these remote cutting activities. Segmentation using abrasive water blast produces finer particles to be controlled than alternative mechanical cutting methods.

Larger pieces of activated metal, typically from recuts with either of the segmentation processes may present extremity and whole-body exposure situations for concern. Smaller “chips,” depending on activation levels, additionally may present skin exposure challenges upon removal of equipment and waste from beneath the water surface. Since more highly activated pieces are more easily located and captured through radiation surveys, the lesser activation chips and fine pieces may become the primary sources of activated metal DRPs for the remainder of decommissioning activities. Containment of DRPs within water is accomplished through cleaning and survey of materials and equipment being removed from the cavity, pool or vessel during segmentation processes. It is critical to control DRPs at the source.

Reactor vessel segmentation presents a different challenge as this activity is performed without the advantage of water to mitigate the spread of these DRPs. Although the activity levels generally are less than internals, the active core region of the reactor vessel may generate significantly activated particles during segmentation. The use of capture mechanisms, forced ventilation flow, and routine monitoring/cleaning of the segmentation equipment/area are additional means to limit DRP spread.

General equipment and work area setup, for both in-vessel and reactor vessel segmentation should include boundary controls that provide for equipment and personnel monitoring prior to leaving the area. In addition, maintaining appropriate equipment and radiological boundary margins, performing routine area cleaning, and frequent change out of access/egress area step-off pads is necessary. Frequent surveys of the control boundary are also essential.

To the extent possible, packaging of waste and used equipment should be done within the controlled area. In instances where that is not possible, the use of fixatives prior to removal from the area is preferred. The fixatives may be applied to minimize the spread of contamination and particularly DRPs. Exterior temporary enclosures may be warranted for extensive removal and packaging of material outside of containment or other building structure.

H.2.3 Activated Concrete

For areas of activated concrete (bio-shield wall, under vessel concrete, etc.), demolition should be performed inside of containment. Dust control is maintained through water misting and should be applied along with general area forced negative ventilation. Plans for excessive water (i.e., runoff from misting) are required to prevent contamination from spreading due to the water runoff. Ventilation should provide air flow at the location of work activity and away from personnel.

Control of activated concrete demolition activities often will result in a large area of containment to be a “controlled area.” This is where packaging of the demolition debris and waste will occur. Protective wrapping of package surfaces may provide for quicker decontamination/clearance at time of release. An area outside the boundary should provide for monitoring of personnel, equipment and package waste coming out of the area. Exterior enclosures may be warranted for extensive removal and packaging of material outside of containment. The levels of radioactivity from activated concrete are generally much lower than compared to fuel and activated metal as the neutron activation is lower in these areas containing concrete.

H.3 DRP Surveys

H.3.1 Detection Capability

Perform predetermined periodic surveys for DRPs around established boundaries to locally control the areas where DRPs are being generated. Use of large area smears and identification of lower activity particles may provide a precursor to a larger problem.

The Oak Ridge Institute for Science and Education (ORISE) document “Estimating Scan Minimum Detectable Activities of Discrete Radioactive Particles,” October 2022 (ML22304A137) identifies issues with the detection of DRPs during normal final status surveys of land areas. The document states, that in general, the scan MDA calculation identifies that the lowest scan MDAs occur when the detector is positioned closest to the ground, the surveyor walks as slow as possible, the DRP is positioned on the surface, and the detector passes directly above the DRP. When planning a final status survey, the survey designer should consider surveyor velocity, ground (or source)-to-detector distance, and potential soil-cover depth. A surveyor velocity of 0.25 m/s may be unreasonable in real-world applications, i.e., surface terrain prevents the surveyor from traversing this slowly. However, optimization of the survey design may include a scan MDA based on a surveyor velocity of 0.25 m/s for small areas receiving follow-up investigations. Similarly, a reasonable ground-to-detector distance may be established for routine scanning that enables the surveyor to avoid hitting the detector on surface debris.

This source-to-detector distance may be reduced in areas requiring a higher level of scrutiny. Because the thickness of soil cover greatly influences the scan MDA, DRP investigation surveys should occur prior to any site actions that have the potential to re-distribute DRPs into deeper soil strata.

H.3.2 Advanced Instrumentation

Advanced digital spectrometry is available to better record and resolve interactions for various types of instrument detector crystals. The advanced systems are designed to be capable of multiple inputs from an array of detectors (see example RS-700, Mobile Radiation Monitoring System). Sizes and types of crystal material should be chosen to optimize the detection capabilities and depend upon the radiological isotopes of interest. Due to the size and weight of these detection systems, they are often best suited for surveys of large areas, using utility terrain vehicles (UTV) or tractors with GPS logging technology.

Below are specific examples of advanced instrumentation that can be applied to land surveys for both DRPs and distributed residual radioactivity. In these cases, radiological raw data is logged along with the detector global position coordinates and time correlated for post-processing. For DRPs, there are likely many options for data assessment in the post-processing phase that may need to be examined and discussed with the stakeholders. However, without a discrete particle detection activity threshold (i.e. action level, detection level etc.) and a reasonable source-to-detector geometry, the resulting sensitivity is difficult to assess against a standard.

- In one case, a survey system was employed that consisted of six 2”x2” NaI(Tl) detectors. The detectors were placed in a 62-inch-wide array and mounted on a UTV. The array was wired to a 12-channel counter, data logger and a single GPS receiver and antenna combined with a high-accuracy inertial measurement unit (IMU). The GPS receiver, and the IMU are integrated using an on-board tablet or laptop computer (control computer) running scanning software. In this case, detector collimators were not used since a wide viewing angle was desired for the detection of DRPs.

- The survey process involved advancing the UTV at a speed of 0.5 mph and then uploading data sets for post-processing on at least a daily frequency.

- The post processing was performed using CAD/GIS software. The initial data assessment showed that the detector response across the scan surface varied significantly. This result was due to variations of NORM radioactivity. To overcome this condition, the software used a fishnet grid analysis where each grid represented an area of 20 ft square. The software then analyzed the data in each grid for statistics (min, max, mean and Z-score). This was coupled with color coding of the data the on map by z-scores. This permitted identification of elevated areas (or discrete locations) for further analysis.

- The sensitivity of this system was established by performing a detailed efficiency spatial calibration coupled to a probabilistic MDA analysis.

- Another tool that can be used is a smaller version of the 6-detector system described above. This system uses a single detector operated in a traditional scan mode using a surveyor with data logging using GPS. In this case, the surveyor manually walks a defined area and moves the detector laterally. The detector data is logged along with the GPS/time data for post processing where an analysis, similar to the 6-detector array, can be performed.

- For smaller areas, newly designed detector crystals for surface area scans are available that will increase the probability of the detector passing directly over a DRP with use of a handheld instrument. To minimize the weight for surveys with handheld detectors, plastic scintillators can replace the heavier NaI crystals. (i.e., Ludlum 44-132). By increasing the volume of plastic scintillator, the same distributed activity detection capability can be obtained as the standard 2”x2” NaI while increasing the probability of DRP detection.

For building surfaces, paved or concrete surface areas, surveys typically consist of both scanning and fixed measurements for beta and/or alpha radiation. Some advanced technologies rely on data logging and positional tracking such that the data can be aggregated in a manner similar to the land survey methods discussed above. As robotic technologies advance, the methods used to collect this data and control the scan speed will be more available at lower cost. One technology uses a position sensitive proportional counter that can identify the location of the source of radiation along a detector anode length of 1 to 2 meters. Control of scan speed and tracking coincident is essential with data logging. In these applications, data can be collected in $100cm^2$ areas, represented by 100 measurement areas for each 1 m^2.

H.4 Final Status Surveys and DRPs

Areas that have or have been remediated for DRP contamination spread should have enhanced survey performed for DRPs. If DRPs are found during the FSS, then the DRPs shall be removed, an extent of condition performed (see Dose Assessment and Safety Significance Below), and the particle disposed of properly.

H.4.1 DRP Dose Assessment and Safety Significance

DRPs trapped in the upper respiratory system are either quickly cleared to the GI tract or to the environment. Although there is no specific dose limit for DRPs, VARSKIN may now be used to assess the effects to the upper respiratory tract, and both small and large intestines. Additionally, VARSKIN may aid in the identification of the potential for ulceration and cancer causation to the organs evaluated.

H.4.2 VARSKIN Use for Ingestion

Discrete Radioactive Particle (DRPs) have been a radiological concern and dosimetry challenge over the last few decades, especially in and around nuclear power generation facilities. During this time, VARSKIN has been used to calculate the exposure to skin resulting from contamination from hot particles. VARSKIN was benchmarked against Monte Carlo N-Particle (MCNP) software simulations, to evaluate its ability to be used for the calculation of beta doses to the digestive tracts in the case of hot particles ingestion.

VARSKIN was found to be in alignment with the calculation of the maximum dose from ingested hot particles. The VARSKIN code results were found to be within approximately 10 percent of those from MCNP, for electron energies between 0.2 to 2.5 MeV and particle sizes less than a few hundred micrometers in diameter.

However, to perform such calculations in VARSKIN, it was found that a few enhanced parameters must be included in the calculation. The first is to cancel the backscatter correction. Second, an appropriate volume averaging parameter for the organ model must air be included. This should be done according to the International Commission on Radiological Protection (ICRP). Third, the user must set the averaging area to give the maximum Dose Area Product (DAP).

With these enhanced parameters, the dosimetry from VARSKIN will be able to estimate the worst case for the hot particle exposure that mainly relates to the local dose for a potential ulceration risk or the average dose for cancer risk.

For a hypothetical exposure, dose distribution around a cylindrical brachytherapy source inside the body was also calculated using VARSKIN. VARSKIN results compared well to MCNP version 6.2 and the Electron Gamma Shower (EGSnrc) software package when a point source was modeled without selfattenuation and with the source at distances more than approximately 1 millimeter (mm) away. When realistic source composition was included in the model, VARSKIN produced results that were approximately 30 percent lower than those from EGSnrc.

See **Using VARSKIN for Hot Particles Ingestion Dosimetry Evaluation**, NUREG/IA-0535 (ML22255A157) for further information.

H.4.3 Dose Coefficients for DRPs

For a DRP that is inhaled or ingested, theoretical doses may be determined for a given activity particle through use of dose coefficients. These values can be found in Dose Coefficients for Discrete Radioactive Particles (DRP), presentation by David M. Hamby, PhD (ML22305A584).

The presentation discusses recommended ulceration threshold for internal DRPs and dose coefficients for stationary DRPs for skin surface, upper respiratory tract, and intestines (small and large). In some of these cases, the dose is averaged over an area well less than 10 cm^2 so may not be directly comparable to skin doses from DRPs.

APPENDIX I. EXAMPLES OF WORK PERFORMED AT RISK BEFORE THE LTP IS APPROVED

I.1 Humboldt Bay Power Plant Unit 3 At Risk Work

Substantial excavations and backfill were required prior to LTP draft completion. The following work took place three to five years before the LTP was submitted to the NRC for approval.

I.1.1 Humboldt Bay Generating Station (HBGS) -Fossil Unit

Preconstruction activities for HBGS, a new generation adjacent to HBPP Unit 3, included removal of all underground commodities from the proposed location of the facility. Although the area had been slightly impacted by HBPP Unit 3 (the nuclear unit) operation, the most significant issue was related to a stormwater drain line. This drain line had been impacted from a previously overflowed radioactive concentrated liquid waste tank.

Prior to staffing for decommissioning, the contracts for remediation activities and plans for survey requirements were deficient due to a lack of MARSSIM and decommissioning experience of licensee personnel. Near the end of the remediation efforts, an NRC ORAU survey site visit exposed these deficiencies. As a result, significant evaluations were required to assimilate the existing radiological data of the as-left conditions necessary to justify that the area met the clearance criteria.

I.1.2 New Switchyard for HBGS

As the HBPP Unit 3 decommissioning organization was formed, the project management presented plans to construct a new electrical switchyard over the Unit 3 cooling water intake lines from the intake canal. For structural integrity of the new facilities, the intake lines needed to be removed. The intake lines were considered impacted due to historical spills into the canal. This resulted in a large excavation and backfill. Generic DCGL values were utilized for excavated area surveys prior to backfill. Backfill was completed with soil cleared by MARSSIM based random sampling methods.

I.2 Million-Gallon Fuel Oil Tank Area Environmental Remediation

A **one-million-gallon** fuel oil tank within a soil berm area was demolished and the residual fuel oil contamination was remediated. Prior to backfill of the remediated area, surveys and soil sampling were performed to the generic DCGLs. Work plans for the surveys were written in an attempt to satisfy MARSSIM guidance without an approved LTP and specific MARSSIM survey procedures. Because there was no LTP guidance for backfill soil at that time, clearance of the soils required multiple samples prior to backfill soil placement.

I.3 Other Considerations

At many power plants being decommissioned, the turbine building at pressurized water reactors and other buildings outside the Radiological Control Area (RCA) have been surveyed, demolished and backfilled before the LTP was approved. As a result, this work has typically been performed at risk. It is suggested that the licensee request NRC have ORISE perform verification survey of these areas directly after the licensee surveys have been performed and request that the NRC review the FSS reports of these areas first.

After the surveys of these areas are completed, the areas could then be fenced off and placed under isolation controls. Any changes from the draft LTP to the approved LTP that affect these areas would require a revision to the FSS reports. The probability of a need to resurvey these areas should be low unless (1) there was inadequate contamination control for these areas; (2) there was a significant change to survey or dose calculation methods; or there existed a groundwater plume affecting these areas.

There may be a possibility that the approved LTP could impact these surveys and result in changes to the evaluation or the development of a re-sampling strategy to meet LTP requirements with NRC concurrence. The impacts could be a result of the following:

- Changes in the ROCs or IC Dose fractions,
- Changes in the allocation of dose fractions that could change the operational DCGLs (see Appendix B), or
- Changes in dose modeling assumptions resulting in changes to the base case and operational DCGLs.

APPENDIX J. MARSSIM CHEAT SHEET

PTP's MARSSIM CHEAT SHEET

MARSSIM OVERVIEW

- MARSSIM (NUREG-1575) is a document that provides guidance for conducting final status surveys at radiologically contaminated facilities undergoing decommissioning.
- The final status survey is conducted by the licensee (or a subcontractor) after they have concluded that further remedial action is not necessary. In other words, the site should be relatively clean and is expected to meet the release criterion established by the regulator.
- A single final status survey is not conducted over the entire site. Instead, final status surveys are conducted in discrete areas of the site known as survey units. The planning, implementation and data assessment for the final status surveys proceed independently for each survey unit. Ultimately, every survey unit at a site must be demonstrated to meet the regulator's release criterion.
- The Nuclear Regulatory Commission's criterion for unrestricted release of a property can be summarized as follows: residual contamination that is distinguishable from background should not result in more than 25 mrem in a single year to an average member of the critical population. Many individual states employ lower criteria (e.g., 10 mrem), but we will assume for the purpose of this discussion that the criterion is 25 mrem.
- The concentrations (typically in soil or on building surfaces) that result in 25 mrem are referred to as derived concentration guideline levels (DCGLs).

IMPORTANT GUIDANCE DOCUMENTS

Abelquist, E. Decommissioning Health Physics. A Handbook for MARSSIM Users. Second Edition. CRC Press, Taylor & Francis. 2014.

NUREG-1505. Nuclear Regulatory Commission. A Nonparametric Statistical Methodology for the Design and Analysis of Final Status Decommissioning Surveys. Revision 1. 1998.

NUREG-1507. Nuclear Regulatory Commission. Minimum Detectable Concentrations with Typical Radiation Survey Instruments for Various Contaminants and Field Conditions (pre-published draft). June 1998.

NUREG-1575. Multi-Agency Radiation Site Survey and Investigation Manual (MARSSIM). Revision 1. August 2000.

NUREG-1576. Multi-Agency Radiological Laboratory Analytical Protocols Manual. July 2004.

NUREG-1757 (three volumes). Nuclear Regulatory Commission. Consolidated Decommissioning Guidance.

DATA QUALITY OBJECTIVES (DQO) PROCESS (MARSSIM Appendix D)

The seven step DQO process is an integral component of the planning phase of the data life cycle. Its primary purpose is to ensure that all the important issues are addressed. These seven steps can be boiled down to the following:

1. State the problem: Identify the planning team, decision makers, deadlines, resources and a concise description of the problem.
2. Identify the decision: for a final status survey this would be "Is the level of residual contamination in a given survey unit below the release criterion." Then, the alternative actions are identified e.g., further remediation, reevaluation of the DCGLs, restrictions on release, etc.
3. Identify inputs to the decision: Identify the specific questions to be answered, e.g., "What physical characteristics of the site need to be evaluated," "What chemical characteristics of the contamination need to be determined," etc. The chosen means to answer these questions are identified. The information needed to establish the DGCLs is identified. What methods will be used to provide the necessary data is determined.
4. Define the study boundaries: Areas of the site to be evaluated are defined, and the time frame in which the survey will be performed is defined.
5. Develop a decision rule: the statistical method for describing the residual activity is identified e.g., the mean, median for the survey unit, etc. The action levels are identified.
6. Specify limits on decision errors: Estimate the likely variation in the measurements for the survey unit, identify the null hypothesis and define the consequences of Type I and Type II errors in terms of health, political, and resource issues. Specify acceptable values for Type I and II error rates (alpha and beta). The formal DQO process does not reference the LBGR, but the latter should be specified when beta is specified.
7. Optimize the design of the survey for obtaining the data. Evaluate data collection design alternatives, develop the mathematical expressions that will be necessary to implement the alternatives and select the optimal options.

MARSSIM DATA LIFE CYCLE (MARSSIM 2.3)

In all four phases of the data life cycle, communication between the licensee and the regulator is essential.

1. Planning Phase (design the survey)	**2. Implementation Phase** (perform the survey)
3. Assessment Phase (evaluate the measurements)	**4. Decision Making Phase** (what to do if the survey unit fails to meet the release criterion)

1. PLANNING PHASE OF THE DATA LIFE CYCLE

1. Determine the $DCGL_w$ for Individual Nuclides (outside scope of MARSSIM)

- It might be possible to use screening levels published by the NRC (NUREG-1757 Vol 2.) with minimal justification.
- If a screening level is not available for a particular radionuclide, one might be calculated using default input parameters and the DandD code.
- A less conservative (i.e., higher) DCGL might be calculated using site-specific input parameters and the computer code RESRAD or RESRAD-Build.

2. Determine the Gross $DCGL_w$ for Multiple Nuclides when Performing Gross Alpha or Beta Measurements (MARSSIM 4.3.4)

- Use the most restrictive (lowest) of the DCGLs for individual nuclides, or
- Determine the fraction of the total activity (alpha or beta) contributed by the various nuclides and use the following equation:

$$DCGL_{GROSS} = \frac{1}{\frac{f_1}{DCGL_1} + \frac{f_2}{DCGL_2} + \dots \frac{f_n}{DCGL_n}}$$

$DCGL_{GROSS}$ is the gross alpha or beta activity DCGL for the specified mix of nuclides.
f_1, f_2 , etc. are the fractions of the total alpha or beta activity contributed by nuclide 1, nuclide 2, etc.
$DCGL_1$, $DCGL_2$, etc. are the individual DCGLs for nuclide 1, nuclide 2, etc.

3. Adjust (lower) the $DCGL_w$ for Surrogate Nuclides (MARSSIM 4.3.2)

It is possible to use measurements of a nuclide that is inexpensive to analyze (e.g., Cs-137 by gamma spec) as a surrogate for the measurements of one or more expensive to analyze nuclides (e.g., Sr-90 by radiochemistry) if a ratio can be established between the nuclides. When this is done, the DCGL for the nuclide that is measured (the surrogate) must be adjusted downwards.

$$DCGL_{ADJ\ SURR} = \frac{1}{\frac{1}{DCGL_{SURR}} + \frac{R_2}{DCGL_2} + \dots \frac{R_n}{DCGL_n}}$$

$DCGL_{ADJ\ SURR}$ is the adjusted (lowered) DCGL for the surrogate nuclide.
R_2, etc. are the expected ratios of the activities of the non-detected nuclides to the activity of the surrogate nuclide.
$DCGL_2$, etc. are the individual DCGLs for nuclide 2, etc.

4. Determine the $DCGL_{EMC}$ (NUREG-1505 Chapter 8)

The $DCGL_{EMC}$ is the maximum permitted average concentration in a hot spot. It is the concentration of a specified nuclide in a specified area (smaller than the survey unit) that is assumed to result in 25 mrem in a year (i.e., the release criterion). It is calculated as follows:

$DCGL_{EMC} = DCGL_W \times AF$ **AF** is an area factor that is specific to the nuclide and area.

Although tables of example area factors have been published, there are no default area factors (or $DCGL_{EMC}$ for that matter). To calculate the area factor, divide the dose predicted with RESRAD (or RESRAD-Build) for the survey unit (or default area) by the dose predicted for the area of the hot spot. In the planning phase of the data life cycle (e.g., step 15), a worst case hot spot size must be assumed. This assumed area is that bounded by four measurement/sampling points. During data assessment, actual hot spot areas are used to determine the $DCGL_{EMC}$.

5. Classify the Site According to Contamination Potential (MARSSIM 4.4, NUREG-1505 2.2.3, 2.2.4)

Each area of the site is assigned one of the following classifications according to its potential for contamination

- Class 1 impacted areas have or had a potential for individual measurements above the DCGL. Remediated areas generally considered Class 1.
- Class 2 impacted areas have or had a potential for contamination at a significant fraction of the DCGL. Individual measurements should not exceed the DCGL.
- Class 3 impacted areas have little or no potential for contamination. Individual measurements should not exceed a significant fraction (e.g., 10-20%) of the DCGL.
- Non-impacted areas have no potential for contamination. If possible, reference area(s) are established in non-impacted area. Note that the definitions of impacted areas in NUREG-1757 are more flexible than the above which are given in MARSSIM.

6. Establish the Survey Units (MARSSIM 4.6)

The site is divided up into areas of similar contamination potential known as survey units. The survey unit is the fundamental unit of compliance. Planning, implementation and data assessment are conducted independently for each survey unit. Maximum recommended survey unit areas:

	Structure (e.g., building)	Land Area
Class 1	100 m^2 floor area	2000 m^2
Class 2	1000 m^2 floor area	10,000 m^2
Class 3	no limit	no limit

7. Determine if Scenario A or B will be Used

- Scenario A is the most commonly employed approach in final status surveys, and the only approach described in MARSSIM (NUREG-1575). The object is to demonstrate that the average/median level of residual radioactivity in a survey unit is less than the DCGL.
- In Scenario B, the object is to demonstrate that measurements in the survey unit are indistinguishable from those in background. Scenario B is used when the DCGL is low, the radionuclide is in background (or the measurements are not nuclide specific) and background is variable. The primary source of guidance for Scenario B is NUREG-1505.

8. Determine if the Sign Test or Wilcoxon Rank Sum (WRS) Test will be Used to Assess the Data

- Analysis is nuclide-specific and the nuclide is not in background (e.g., Co-60 in soil) — use the Sign test.
- Analysis is nuclide-specific and the nuclide is in background (e.g., Pu-239 in soil) at a small fraction of the DCGL — use the Sign test.
- Analysis is nuclide-specific, and the nuclide is in background (e.g., Ra-226 in soil) at a significant fraction of the DCGL — use the WRS test.
- Analysis is not nuclide-specific (e.g., gross beta on surfaces) and background is a small fraction of the DCGL — use the Sign test.
- Analysis is not nuclide-specific (e.g., gross alpha on surfaces) and background is a significant fraction of the DCGL. This situation can be problematic if surface types and background vary from one survey unit to another. In this case, there are several options to account for background (Chapter 12 NUREG-1505):
  - — use the WRS test, or
  - — subtract average background for surface types from survey unit gross measurements and use Sign test on net values, or
  - — subtract paired background measurements for surface types from gross measurements and use Sign test on net values.

9. Determine if the Unity Rule will be used in the Statistical Tests (NUREG-1505 Chapter 11)

The unity rule is used when two or more nuclides are analyzed in each soil sample (or alpha and beta measurements are performed at each location). The unity rule can be thought of as a sum of the fractions approach wherein the “concentrations” of multiple radionuclides are expressed as fractions of the DCGLs. When a DCGL for multiple nuclides is used in these tests, it is assigned a value of 1.

$$\text{Combined concentration of multiple nuclides} = \frac{\text{Conc. nuclide 1}}{DCGL_1} + \frac{\text{Conc. nuclide 2}}{DCGL_2} + \text{etc.}$$

1. PLANNING PHASE OF THE DATA LIFE CYCLE continued

10. Select the Type of Detection Equipment (MARSSIM Chapter 6)

- Performing scans and static measurements on structural surfaces — gas flow proportional counters usually preferred.
- Performing scans for gamma emitters in soil — sodium iodide detectors usually preferred.
- Measuring contaminants in soil — collect soil samples and analyze by gamma spectroscopy and/or radiochemistry. Contaminants in soil might also be measured by in-situ gamma spectroscopy.

11. Determine the Measurement Protocols (MARSSIM 5.5.3)

- Class 1 Survey Units — Scan coverage: 100%. Measurements/samples collected in systematic (e.g., triangular) pattern.
- Class 2 Survey Units — Scan coverage: 10 to 100%. Measurements/samples collected in systematic (e.g., triangular) pattern.
- Class 3 Survey Units — Scan coverage: judgmental. Measurements/samples distributed randomly.
- Non-impacted area — No scan or measurements required.

12. Determine the Measurement and Scan MDCs (MARSSIM 6.7, NUREG-1507 3.1 and Chapter 6). One of the few absolute requirements in MARSSIM is that the measurement MDC be below the $DCGL_W$. Nevertheless, MARSSIM recommends a MDC that is 10-50% of the $DCGL_W$. Typical measurement MDCs for gas flow proportional counters found in NUREG-1507 Table 5.2. Example lab sensitivities found in MARSSIM Table 7.3.

$$\text{Measurement MDC (dpm/100cm}^2) = \frac{3 + 4.65\sqrt{C_B}}{t\ E_i\ E_s\ \frac{A}{100}}$$

C_B is the background count (counts).
t is the count time (min). This equation assumes background and sample count times are identical.
E_i is the instrument (2 pi) efficiency (counts per particle leaving the surface).
E_s is the surface efficiency (fraction of the decays that a detectable particle leaves the surface). Default is 0.5 for betas with maximum energies above 400 keV and 0.25 for alphas and betas with maximum energies between 150 and 400 keV.
A is the probe area (cm^2).

Typical NaI scan MDCs can be found in Table 6.7 of MARSSIM and Table 6.4 of NUREG-1507.

$$\text{Scan MDC (dpm/100cm}^2) = \frac{60\ d'\ \sqrt{C_{Bi}}}{i\ \sqrt{p}\ \ E_i\ E_s\ \frac{A}{100}}$$

d' is 2.32 if the acceptable probability of false positives is 0.25 and the acceptable probability of a correct detection is 0.95 (see MARSSIM Table 6.5).
C_{Bi} is the background count during the time interval i.
i is the time interval (sec) that the probe is over the hot spot (of an assumed size).
p is the surveyor efficiency (MARSSIM recommends 0.5 be used).

13. Determine the Scan and Measurement Investigation Levels (MARSSIM 5.5.2.6)

The investigation level is the instrument response (e.g., cpm) that triggers an investigation when exceeded. MARSSIM suggests the following:

- Class 1 survey unit. The instrument response corresponding to the $DCGL_{EMC}$ for the area bounded by four measurement points.
- Class 2 survey unit. The instrument response corresponding to the $DCGL_w$. If this is exceeded, the survey unit might have been misclassified.
- Class 3 survey unit. The instrument response corresponding to some fraction of the $DCGL_w$. If the scan MDC exceeds the $DCGL_w$, the instrument response at the scan MDC might be used.

14. Determine the Acceptability of Type I and Type II Errors and Set the LBGR (MARSSIM Appendix D, NUREG-1505 Chapter 2)

For the purpose of the statistical tests, the Null (working) Hypothesis is that the median level of contamination in the survey unit exceeds the $DCGL_w$.

- Regulator establishes maximum acceptable probability (alpha) of the statistical test falsely concluding that the median level of contamination (above background) in the survey unit is below the $DCGL_w$ when it is actually at or above it. The most likely value for alpha is 0.05 (i.e., 5%).
- Licensee establishes acceptable probability (beta) of the statistical test falsely concluding that the median level of contamination (above background) exceeds the $DCGL_w$ when it is at a concentration known as the lower boundary of the gray region (LBGR).
- The licensee sets the LBGR at some concentration below the $DCGL_w$. In general, the LBGR should be set at the expected average/median concentration in the survey unit.

15. Determine the Appropriate Number of Measurements or Samples (MARSSIM 5.5.2, NUREG-1505 Chapter 9)

Calculate the relative shift. Given the relative shift and values for alpha and beta, the required number of measurements or samples can then be found in Table 5.3 of MARSSIM if the WRS test is to be used. If the Sign test is to be used, the number of measurements is found in Table 5.5.

$$\text{Relative Shift} = \frac{DCGL_W - LBGR}{\sigma}$$

The relative shift is a unitless number (often between 1 and 4) related to the chance that individual measurements will exceed the $DCGL_w$. The smaller the relative shift, the greater the likelihood some measurements exceed the $DCGL_w$ and the greater the number of measurements that should be made.
σ is the expected variability of the measurements. It, like the LBGR, is based on earlier characterizations.

If the unity rule is employed:

- use 1 as the value for the DCGL in the relative shift calculation, and
- use the following equations to determine the appropriate values for the LBGR and σ.

$$LBGR = \frac{\text{Expected conc. nuclide 1}}{DCGL_1} + \frac{\text{Expected conc. nuclide 2}}{DCGL_2} + \text{etc.} \qquad \sigma = \sqrt{\left(\frac{\sigma}{DCGL}\right)^2_{\text{nuclide 1}} + \left(\frac{\sigma}{DCGL}\right)^2_{\text{nuclide 2}} + \text{etc.}}$$

16. For Class 1 Survey Units, the Number of Measurements might need to be Increased (MARSSIM 5.5.2.4)

This is because the scan must be sufficiently sensitive to detect a hot spot exceeding its $DCGL_{EMC}$. The largest (worst case) potential hot spot area is assumed to be that bounded by four measurement points (the survey unit area divided by the number of measurements/samples). If the scan MDC is below the $DCGL_w$, the scan MDC is also below the $DCGL_{EMC}$ and there is no problem. If the scan MDC is above the $DCGL_w$, it must be compared with the $DCGL_{EMC}$ for a hot spot of that area. Then, if the scan MDC is above the $DCGL_{EMC}$, the number of fixed measurements/samples must be increased so that the increased $DCGL_{EMC}$ equals the scan MDC. To do this, we first divide the actual scan MDC by the $DCGL_W$. This gives the area factor for the new smaller hot spot where the scan MDC equals the $DCGL_{EMC}$. Then the hot spot area corresponding to this area factor is determined. Dividing this new area into the total survey unit area gives the new required number of measurements/samples.

17. Establish Reference Grid and Determine Measurement/Sample Locations (MARSSIM 5.5.2.5, NUREG-1505 Chapter 3.5)

MARSSIM does not recommend a particular type of reference grid. When measurements/samples are to be distributed in a systematic pattern (Class 1 and 2 survey units), MARSSIM recommends a triangular (equilateral) pattern. The reference grid coordinates of the starting point for a systematic pattern are determined using random numbers. If the measurements/samples are to be distributed randomly (class 3 survey units), the coordinates for all the locations are selected using random numbers.

2. IMPLEMENTATION PHASE OF THE DATA LIFE CYCLE

1. Scan Surfaces for Contamination (MARSSIM 6.4.2)

The detector probe is slowly moved back and forth over potentially contaminated surfaces while the surveyor listens to the detector's audio output for indications that the investigation levels have been exceeded. The primary purpose of the scan is to locate small areas of elevated activity, i.e., hot spots. If located, the latter are characterized by additional measurements/samples to determine that the contamination is below the $DCGL_{EMC}$.

- For alpha and beta scans, the probe is usually 0.5 to 1.0 cm above the surface. A typical scan rate might be one half to one probe width per second.
- For gamma scans, a NaI probe is swung over the ground in a 1 m arc approximately 10 to 15 cm above the surface. A typical scan rate is 0.5 m/s.

2. Perform Static Measurements on Surfaces and/or Collect Soil Samples (MARSSIM 6.4.1, 7.5)

This is done to obtain accurate determinations of the contamination levels at a number of unbiased locations. This data will be assessed statistically to determine if the contamination levels in the survey unit are below the $DCGL_w$ and used to determine that the survey unit was properly classified.

- Static measurements of alpha or beta concentrations are performed with the probe directly on, or just above, the surface (in the final status survey, the surfaces should be clean with little to no removable activity). Measurements are typically performed using scalars set to one minute count times.

$$C = \frac{R_N}{E_i\, E_s\, \frac{A}{100}}$$

C is the surface concentration (dpm/100 cm²).
R_N is the net count rate (cpm).
E_i is the instrument efficiency (2 pi efficiency).
E_s is the surface efficiency.
A is the probe area (cm²).

Default values for surface efficiency (E_s):
- 0.5 for betas with maximum energies above 400 keV.
- 0.25 for alphas and betas with maximum energies between 150 and 400 keV.

- Soil samples are generally collected to a depth of 15 cm. One kilogram samples are usually collected for gamma spec analysis, while smaller samples might be obtained if a radiochemical analysis is performed.

3. ASSESSMENT PHASE OF THE DATA LIFE CYCLE

1. Data Verification (MARSSIM 9.3.1, NUREG-1576 Chapter 8)

It is determined whether or not the laboratory and field personnel did what they were supposed to do. For example:
- were the correct instruments used and were daily QC checks on the instruments performed.
- were the count times and sample masses what they were supposed to be (i.e., were the requisite MDCs obtained).
- were the requisite number of spilt, duplicate, blank samples performed.
- is there missing documentation.

2. Data Validation (MARSSIM 9.3.2, App. N, NUREG-1576 Chapter 8)

Data points are assessed and flagged as necessary. In essence, this is a reality check on individual measurements. Typical qualifiers (flags) are:

• U measurement less than critical level. • J measurement very uncertain or questionable. • R data point rejected.

3. Preliminary Data Review (MARSSIM 8.2.2)

- Determine the following: number of valid measurements, lowest measurement, highest measurement, mean, median, and standard deviation.
- Based on the measurements and scan data, determine if the area classification appears correct.
- Determine that the requisite number of measurements are made.
- Determine if the mean is above or below the LBGR.
- Survey unit fails if the average measurement (Sign test) or the difference between the average survey unit and reference area measurements (WRS test) exceeds the $DCGL_w$. Note, this does not include biased measurements, only those collected for the purpose of the statistical tests.
- A statistical test is not necessary if all the measurements (Sign test), or the difference between the highest survey unit measurement minus the lowest background measurement (WRS test), is below the $DCGL_w$. As before, this does not include judgmental and biased measurements.
- A statistical test is necessary if the average measurement (Sign test) or the difference between the average survey unit and reference area measurements (WRS test) is below the $DCGL_w$ but some measurements are above the $DCGL_w$. This doesn't include biased measurements.

4. Data are Plotted/Graphed (MARSSIM 8.4, NUREG-1505 4.2.2)

- A posting plot is produced in which the measurements (e.g., pCi/g or dpm/100 cm²) are indicated on a drawing of the survey unit at the locations where the measurements were taken.
- A histogram might also be generated.

5. If necessary, the Sign Test is Performed (MARSSIM 8.3, NUREG-1505 Chapter 5)

This test only employs the unbiased randomly distributed or systematic measurements. Judgmental or otherwise biased data are not used.
- The total number of measurements being evaluated in the survey unit is N. The number below the $DCGL_w$ is the statistic S. If a measurement is the same as the $DCGL_w$, it is not counted and the total number of measurements (N) is reduced by one.
- If S is above the appropriate critical value in Table I.3 of Appendix I in MARSSIM, the Null Hypothesis (that the survey unit exceeds the release criterion) is rejected.
- If S is tied with or below the critical value, we fail to reject the Null Hypothesis and a decision must be made as to how to proceed.

6. If necessary, the Wilcoxon Rank Sum (WRS) Test is Performed (MARSSIM 8.4, NUREG-1505 Chapter 6)

This test only employs the unbiased randomly distributed or systematic measurements. Judgmental or otherwise biased data are not used.
- Add the $DCGL_w$ to each of the reference area measurements.
- The adjusted reference area measurements are then pooled with the survey unit measurements.
- The pooled measurements are ranked (sorted) from lowest to highest (1, 2, 3, etc.). Tied values are assigned an average rank (e.g., 2.5).
- The sum of the ranks of the adjusted reference area measurements is the statistic W_r.
- If W_r is above the appropriate critical value in Table I.4 of Appendix I in MARSSIM, the Null Hypothesis (that the survey unit exceeds the release criterion) is rejected.
- If W_r is tied with or below the critical value, we fail to reject the Null Hypothesis and a decision must be made as to how to proceed.

7. Perform an Elevated Measurement Comparison (MARSSIM 8.5.1, NUREG-1505 Chapter 8)

This test involves all the survey unit measurements, i.e., biased and unbiased measurements.
- Every measurement (above background) above the Investigation Level triggers an investigation.
- The investigation involves confirming the measurement and then determining the area, average concentration, and $DCGL_{EMC}$ for each hot spot.

8. Determine that the Total Dose from All Sources is Below the Release Criterion (MARSSIM 8.5.2, NUREG-1505 8.1)

$$\frac{\delta}{DCGL_W} + \frac{(\text{ave.conc.hot spot } 1 - \delta)}{DCGL_{EMC} \text{ for hot spot } 1} + \frac{(\text{ave.conc.hot spot } 2 - \delta)}{DCGL_{EMC} \text{ for hot spot } 2} + \text{etc.} < 1$$

δ is the average concentration in the survey unit determined from unbiased measurements.

❹ DECISION MAKING PHASE OF THE DATA LIFE CYCLE

A decision must be made as to how to proceed if the survey unit was misclassified, failed the elevated measurement comparison, failed the statistical assessment, or the total dose from all radiation sources exceeded the release criterion.

- Measurements in Class 2 or 3 survey units exceeding the $DCGL_W$ indicate that the area might have been misclassified. If so, the area (including nearby survey units) may have to be recharacterized, reclassified, subdivided into smaller survey units, and resurveyed.
- If the average concentration (above background) in a hot spot is determined during the elevated measurement comparison to exceed the $DCGL_{EMC}$ for that hot spot, the hot spot is remediated and resurveyed.
- If the average concentration (above background) in the survey unit exceeds the $DCGL_W$, or the statistical test fails to reject the Null Hypothesis, the entire survey may have to be remediated and resurveyed.
- The statistical test might fail to reject the Null Hypothesis because the test was performed incorrectly. Check to see if this was the case.
- The statistical test might fail to reject the Null Hypothesis because not enough measurements/samples were obtained. If there is reason to believe that this is the case, the regulator might permit one round of "double sampling." Some indications that not enough measurements/samples were taken include an observation that the actual standard deviation of the measurements was greater than that estimated when the relative shift was calculated, and/or the average of the measurements was higher than the LBGR. In double sampling, additional measurements/samples are obtained at randomly selected locations. These measurements are added to the pool of measurements already obtained and the statistical test is redone.
- The statistical test might fail to reject the Null Hypothesis because the reference area measurements were lower than appropriate for the survey unit. Assess the suitability of the reference area.
- The $DCGL_W$ or $DCGL_{EMC}$ might have been too conservative (low). Consider reevaluating the assumptions/parameters in the dose modeling. Note that this is a last resort. Evaluations of the dose modeling should have been done much earlier.
- Consider releasing the site under restricted conditions.

STATISTICAL CALCULATIONS IN SCENARIO B — INDISTINGUISHABLE FROM BACKGROUND (NUREG-1505 Chapter 13)

In Scenario B, two statistical tests are performed for each survey unit: the WRS test and the Quantile test. The Null Hypothesis in these tests is that the survey unit measurements are indistinguishable from background. As such, the goal is to fail to reject the Null Hypothesis in both tests.

1. Perform the Kruskal-Wallis Test

The purpose of this test is to show that significant variability exists in background. This can be considered a justification for employing scenario B. The Null Hypothesis is that no significant variability exists.

- Obtain measurements/samples in at least four reference areas. NUREG-1505 recommends at least 10 measurements in each, whereas NUREG-1757 recommends at least 15.
- Rank the pooled measurements from all the reference areas. Sum the ranks in the individual reference areas.
- Calculate the Kruskal-Wallis statistic (K) using the following equation:

$$K = \frac{12}{N(N+1)}\left(\sum_{i=1}^{k}\frac{R_i^2}{n_i}\right) - 3(N+1)$$

N is the total number of measurements in all the reference areas.
n_i is the number of measurements in a given reference area.
R_i is the sum of the ranks in a given reference area.

- Compare K with critical value in Table 13.1 of NUREG-1505. In this table, k is the number of reference areas (usually 4). NUREG-1505 recommends an alpha of 0.1 whereas NUREG-1757 recommends a value of 0.2.
- If K exceeds the critical value, the null hypothesis is rejected and the conclusion is that there is significant variability in the background data.

2. Determine a Concentration that is Indistinguishable from Background (e.g., 3 ω)

- The concentration that is indistinguishable from background is some multiple of ω. NUREG-1505 and NUREG-1757 both use three as the multiple.

$$\omega = \sqrt{\frac{(s_B^2 - s_W^2)}{n_0}}$$

s_B is the mean square between the reference areas.
s_W is the mean square within the reference areas.
n_0 is related to the number of measurements in the reference areas.

$$s_B^2 = \frac{\sum_{i=1}^{k} n_i(\bar{x}_i)^2 - \left(\sum_{i=1}^{k}\sum_{j=1}^{n} x_{ij}\right)^2 / \sum_{i=1}^{k} n_i}{k-1}$$

$$s_W^2 = \frac{\sum_{i=1}^{k}\sum_{j=1}^{n_i} x_{ij}^2 - \sum_{i=1}^{k} n_i(\bar{x}_i)^2}{\sum_{i=1}^{k}(n_i - 1)}$$

$$n_0 = \frac{N - \frac{1}{N}\sum_{i=1}^{k} n_i^2}{k-1}$$

N is the total number of measurements in all the reference areas.
n_i is the number of measurements in a given reference area.
k is the number of reference areas.
x_i is an individual measurement in reference area i.

3. Perform Wilcoxon Rank Sum Test on the Survey Unit Data (NUREG-1505 6.3)

- Adjust survey unit data by subtracting the concentration that is indistinguishable from background (3ω) from each survey unit measurement.
- Rank the adjusted survey unit data and the unadjusted reference area data.
- Sum the ranks of the adjusted survey unit data. This statistic (W_s) is compared with the critical value in Table A.7 of NUREG-1505 (m and n are the numbers of survey unit and reference area measurements respectively).
- If W_s is less than or equal to the critical value, the null hypothesis is not rejected and the survey unit is assumed indistinguishable from background.

4. Perform Quantile Test on the Survey Unit Data (NUREG-1505 Chapter 7)

- Each rank is identified as being from the survey unit (S) or a reference area (R). These identifications are then sorted in order from the lowest to highest rank.
- The number of the r highest ranks that are from the survey unit is compared with the value k. Values for r and k are indicated in NUREG-1505's Table A.7. Note that the meanings of m and n are the reverse of those in the WRS test.
- If this number is less than k, the Null Hypothesis is not rejected and the survey unit is assumed indistinguishable from background.

Table 5.3 from MARSSIM

Values of N/2 for Use with the Wilcoxon Rank Sum Test.

To achieve the desired DQOs (acceptable rates of Type I and Type II errors), the total number of measurements is N.

N/2 measurements are taken in the survey unit, and N/2 are taken in the reference area.

Δ/σ is the relative shift.

Table 5.3 Values of N/2 for Use with the Wilcoxon Rank Sum Test

	α=0.01					α=0.025					α=0.05					α=0.10					α=0.25				
	β					β					β					β					β				
Δ/σ	0.01	0.025	0.05	0.10	0.25	0.01	0.025	0.05	0.10	0.25	0.01	0.025	0.05	0.10	0.25	0.01	0.025	0.05	0.10	0.25	0.01	0.025	0.05	0.10	0.25
0.1	5452	4627	3972	3278	2268	4627	3870	3273	2646	1748	3972	3273	2726	2157	1355	3278	2646	2157	1655	964	2268	1748	1355	964	459
0.2	1370	1163	998	824	570	1163	973	823	665	440	998	823	685	542	341	824	665	542	416	243	570	440	341	243	116
0.3	614	521	448	370	256	521	436	369	298	197	448	369	307	243	153	370	298	243	187	109	256	197	153	109	52
0.4	350	297	255	211	146	297	248	210	170	112	255	210	175	139	87	211	170	139	106	62	146	112	87	62	30
0.5	227	193	166	137	95	193	162	137	111	73	166	137	114	90	57	137	111	90	69	41	95	73	57	41	20
0.6	161	137	117	97	67	137	114	97	78	52	117	97	81	64	40	97	78	64	49	29	67	52	40	29	14
0.7	121	103	88	73	51	103	86	73	59	39	88	73	61	48	30	73	59	48	37	22	51	39	30	22	11
0.8	95	81	69	57	40	81	68	57	46	31	69	57	48	38	24	57	46	38	29	17	40	31	24	17	8
0.9	77	66	56	47	32	66	55	46	38	25	56	46	39	31	20	47	38	31	24	14	32	25	20	14	7
1.0	64	55	47	39	27	55	46	39	32	21	47	39	32	26	16	39	32	26	20	12	27	21	16	12	6
1.1	55	47	40	33	23	47	39	33	27	18	40	33	28	22	14	33	27	22	17	10	23	18	14	10	5
1.2	48	41	35	29	20	41	34	29	24	16	35	29	24	19	12	29	24	19	15	9	20	16	12	9	4
1.3	43	36	31	26	18	36	30	26	21	14	31	26	22	17	11	26	21	17	13	8	18	14	11	8	4
1.4	38	32	28	23	16	32	27	23	19	13	28	23	19	15	10	23	19	15	12	7	16	13	10	7	4
1.5	35	30	25	21	15	30	25	21	17	11	25	21	18	14	9	21	17	14	11	7	15	11	9	7	3
1.6	32	27	23	19	14	27	23	19	16	11	23	19	16	13	8	19	16	13	10	6	14	11	8	6	3
1.7	30	25	22	18	13	25	21	18	15	10	22	18	15	12	8	18	15	12	9	6	13	10	8	6	3
1.8	28	24	20	17	12	24	20	17	14	9	20	17	14	11	7	17	14	11	9	5	12	9	7	5	3
1.9	26	22	19	16	11	22	19	16	13	9	19	16	13	11	7	16	13	11	8	5	11	9	7	5	3
2.0	25	21	18	15	11	21	18	15	12	8	18	15	13	10	7	15	12	10	8	5	11	8	7	5	3
2.25	22	19	16	14	10	19	16	14	11	8	16	14	11	9	6	14	11	9	7	4	10	8	6	4	2
2.5	21	18	15	13	9	18	16	13	10	7	15	13	11	9	6	13	10	9	7	4	9	7	6	4	2
2.75	20	17	15	12	9	17	14	12	10	7	15	12	10	8	5	12	10	8	6	4	9	7	5	4	2
3.0	19	16	14	12	8	16	14	12	10	6	14	12	10	8	5	12	10	8	6	4	8	6	5	4	2
3.5	18	16	13	11	8	16	13	11	9	6	13	11	9	8	5	11	9	8	6	4	8	6	5	4	2
4.0	18	15	13	11	8	15	13	11	9	6	13	11	9	7	5	11	9	7	6	4	8	6	5	4	2

Table 5.5 from MARSSIM

Values of N for Use with the Sign Test.

To achieve the chosen DQOs (acceptable rates of Type I and Type II errors), N measurements are taken in the survey unit. There are no reference area measurements.

Δ/σ is the relative shift.

Table 5.5 Values of N for Use with the Sign Test

	α=0.01					α=0.025					α=0.05					α=0.10					α=0.25				
	β					β					β					β					β				
Δ/σ	0.01	0.025	0.05	0.10	0.25	0.01	0.025	0.05	0.10	0.25	0.01	0.025	0.05	0.10	0.25	0.01	0.025	0.05	0.10	0.25	0.01	0.025	0.05	0.10	0.25
0.1	4095	3476	2984	2463	1704	3476	2907	2459	1989	1313	2984	2459	2048	1620	1018	2463	1989	1620	1244	725	1704	1313	1018	725	345
0.2	1035	879	754	623	431	879	735	622	503	333	754	622	518	410	258	623	503	410	315	184	431	333	258	184	88
0.3	468	398	341	282	195	398	333	281	227	150	341	281	234	185	117	282	227	185	143	83	195	150	117	83	40
0.4	270	230	197	162	113	230	1921	162	131	87	197	162	136	107	68	162	131	107	82	48	113	87	68	48	23
0.5	178	152	130	107	75	152	126	107	87	58	130	107	89	71	45	107	87	71	54	33	75	58	45	33	16
0.6	129	110	94	77	54	110	92	77	63	42	94	77	65	52	33	77	63	52	40	23	54	42	33	23	11
0.7	99	83	72	59	41	83	70	59	48	33	72	59	50	40	26	59	48	40	30	18	41	33	26	18	9
0.8	80	68	58	48	34	68	57	48	39	26	58	48	40	32	21	48	39	32	24	15	34	26	21	15	8
0.9	66	57	48	40	28	57	47	40	33	22	48	40	34	27	17	40	33	27	21	12	28	22	17	12	6
1.0	57	48	41	34	24	48	40	34	28	18	41	34	29	23	15	34	28	23	18	11	24	18	15	11	5
1.1	50	42	36	30	21	42	35	30	24	17	36	30	26	21	14	30	24	21	16	10	21	17	14	10	5
1.2	45	38	33	27	20	38	32	27	22	15	33	27	23	18	12	27	22	18	15	9	20	15	12	9	5
1.3	41	35	30	26	17	35	29	24	21	14	30	24	21	17	11	26	21	17	14	8	17	14	11	8	4
1.4	38	33	28	23	16	33	27	23	18	12	28	23	20	16	10	23	18	16	12	8	16	12	10	8	4
1.5	35	30	27	22	15	30	26	22	17	12	27	22	18	15	10	22	17	15	11	8	15	12	10	8	4
1.6	34	29	24	21	15	29	24	21	17	11	24	21	17	14	9	21	17	14	11	6	15	11	9	6	4
1.7	33	28	24	20	14	28	23	20	16	11	24	20	17	14	9	20	16	14	10	6	14	11	9	6	4
1.8	32	27	23	20	14	27	22	20	16	11	23	20	16	12	9	20	16	12	10	6	14	11	9	6	4
1.9	30	26	22	18	14	26	22	18	15	10	22	18	16	12	9	18	15	12	10	6	14	10	9	6	4
2.0	29	26	22	18	12	26	21	18	15	10	22	18	15	12	8	18	15	12	10	6	12	10	8	6	3
2.5	28	23	21	17	12	23	20	17	14	10	21	17	15	11	8	17	14	11	9	5	12	10	8	5	3
3.0	27	23	20	17	12	23	20	17	14	9	20	17	14	11	8	17	14	11	9	5	12	9	8	5	3

Critical Value for Wilcoxon Rank Sum Test	$\frac{m}{2}(n+m+1) + z\sqrt{\frac{nm}{12}(n+m+1)}$
Critical Value for Sign Test	$\frac{N}{2} + \frac{z}{2}\sqrt{N}$

m is the number of measurements in the reference area (WRS test)
n is the number of measurements in the survey unit (WRS test)
N is the number of measurements in the survey unit (Sign test)
z = 1.96 when α = 0.025
z = 1.645 when α = 0.05
z = 1.282 when α = 0.1

MARSSIM IS FLEXIBLE – TAKE ADVANTAGE OF IT!

APPENDIX K. EXPERIENCES AT NUCLEAR POWER PLANT SITES CONCERNING THE EFFECT OF KD ON DCGLS

As discussed in Section 6.1.2, an evaluation should be performed to determine if further justification is needed in the selection of Kds for use in dose modeling calculations that determine DCGLs. This appendix will use experiences from power plant sites to define a process to determine Kds following the guidance in DUWP-ISG-02 which states:

> *Therefore, as a starting point, only Kd values for radionuclides that have a potential to lead to doses greater than 0.025 mSv/yr may require additional support, and only if they are found to be risk-significant. For example, if there is little uncertainty in the Kd value, additional support is likely unneeded.*

K.1 Effect of Kd on Unsaturated Soil DCGL

DUWP-ISG-002, Table 3.6 provides a list of radionuclides including some characterized as having a relatively high uncertainty in defining the values for Kd and/or for which a site-specific study is recommended. NEI has performed a review of experiences at US nuclear plants including approved LTPs of five plant sites that used RESRAD with the resident farmer scenario to determine DCGLs for unsaturated soil. The next subsection describes the results that review concerning Kd for the subset of radionuclides discussed in DWUP-ISG-002 concerning:

- Concentrations of these radionuclides in soil and groundwater
- Results of sensitivity analysis for Kd for radionuclides measured in significant quantities and the Kd selected for input into the RESRAD code.

Table K-1 shows data for the information mentioned in these two bullets from the five sites analyzed. This information is used in the discussion of individual radionuclides that follows.

Before discussing individual radionuclides, some general observations will be discussed. Using the results of the RESRAD code from the Connecticut Yankee (CY) LTP (Reference K-1), Figure K-1 shows the cumulative reduction in contaminated zone soil concentration as a function of the Kd used for in the analysis. The radionuclides and associated Kds used at CY were:

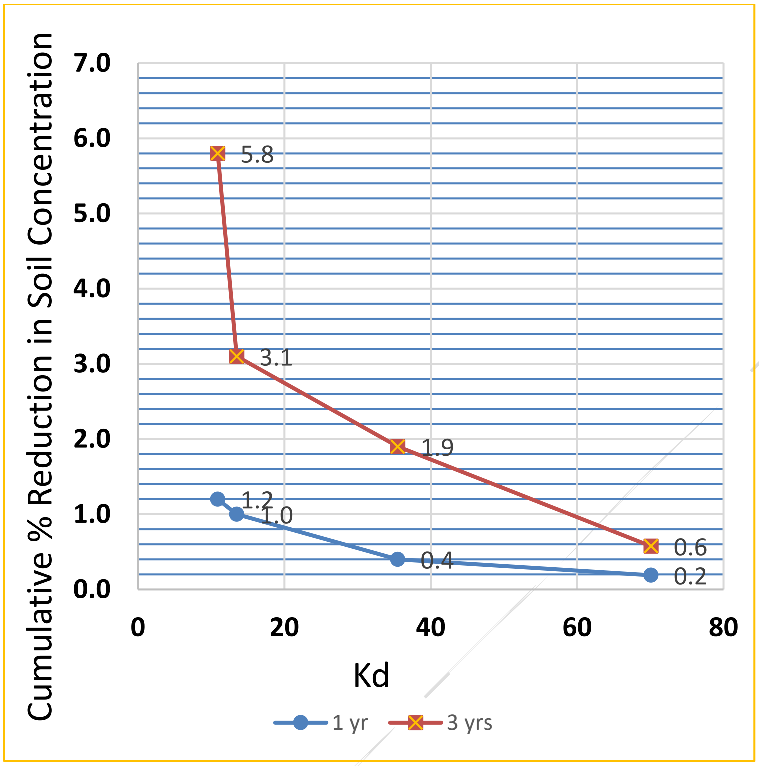
- Sr-90: 10.9
- Tc-99: 13.5
- Ni-63: 35.5
- C-14: 70.1

The graph shows that for a Kd of 13.5, the soil concentration would reduce by 1% in one year and that for a Kd of 50, the reduction is approximately 0.35% in one year. This means that increasing the values of Kd above 13.5 will not increase the DCGL by more than 1%.

Additional review of the CY LTP shows that the peak dose (and correspondingly the lowest DCGL) for these radionuclide (except for Pu-241 as described below) occurs at time zero when all dose is from unsaturated soil related dose pathways. For Pu-241 in the CY analysis, the peak dose occurs at 62 years due to decay to Am-241 but all dose is from unsaturated soil related pathway and therefore was not significantly affected by the Kd used.

This conclusion is further supported by the analysis of the effect of Kd on soil DCGL for the Zion site (Reference K-2). This analysis showed that for the radionuclide (Sr-90) which is the one most affected by Kd, lowering the Kd from 3.4 to 2.3 increased the DCGL by 6% for surface soil and 1 % for subsurface soil. This data shows that in most cases at power plant sites, decreasing the Kd used in the analysis would, if anything, increase the unsaturated soil DCGL. More discussion on this affect is included in the subsections below.

Figure K-1
Cumulative Reduction in Soil Concentration Over time as a Function of Kd



Discussion on Individual Radionuclides

- Am-241
    - This radionuclide has not been detected at significant quantities in soil at any power plant site except Connecticut Yankee (CY). At CY, elevated quantities of Am-241 were measured in the sand bed directly below an outside tank that had experienced long-term leakage. The contamination was on top of the concrete base slab for the tank and did not spread any distance from the slab. The Am-241 contamination was totally remediated as high levels of Co-60 and Cs-137 were also present. Am-241 was not detected in groundwater at the site.
    - Concerning the sensitivity of dose to the Americium Kd, for the four approved LTPs reviewed where sensitivity to Kd was analyzed, it was not sensitive for three cases and showed positive sensitivity (i.e., PRCC>0.25) in one case (Yankee Rowe). The lowest Kd used at any of these sites was 1,250 (Ft Calhoun using median value). These experiences show that to use

very low Kds for Am-241 would likely increase the unsaturated soil DCGL as the parameter is if anything positively correlated to dose.

Table K-1: Detected Radionuclides, Kd Sensitivity Analysis Results and Kds Selected for Unsaturated Soil DCGL Development for 5 Power Plant Sites

Radio-nuclide	Ft Calhoun Only Cs-137 detected in Soil above MDA, Additional ROCs selected based on concrete characterization results (Reference 6-10)			Zion Kd selected based on soil Kd testing at Brookhaven (Reference 5-2)			Rancho Seco (Reference K-3)			Connecticut Yankee - All radionuclides selected for conservatism (Reference K-2)		Yankee Rowe (References K-3 and K-4)		
	Detections in Groundwater – Very low levels of H-3 and Sr-90 detected at 12% of MCL values. (Reference 6-10).			Detections in Groundwater – H-3 and Sr-90 detected near MDC values. Sr-90 detections were upgradient of potential source areas. (Ref. 5-2)			No Detections in Groundwater			H-3, Sr-90 and Cs-137 Detected in Groundwater		H-3 detected in Groundwater		
	Sensitive	Kd selected	Soil DCGL (0.15/1 m) (pCi/g)	Detected in Soil	Kd selected	Surface/ Sub-surface DCGLs (pCi/g)	Detected in Soil	Sensitive Cont. Zone	Kd selected	Sensitive Cont. Zone	Kd selected	Detected as a % of DCGL	Sensitive	Kd selected
Am-241	No	1,250	Deselected	No	N/A	Deselected	No	No	Distrib.	No	1445	<5%	Positive	12,000
C-14	Positive	96.7	57.0 / 9.68	No	N/A	Deselected	Yes	No	Distrib.	Positive	1,970	8.54%	No	Distribution
Fe-55	No	889	Deselected	No	2,857	Deselected	No	No	Distrib.	Positive	7,360	<5%	No	Distribution
Co-60	Positive	5,050	3.77 / 2.93	0.24 pCi/g	1,161	4.26/3.44	Yes	No	Distrib.	No	235	Yes	No	Distribution
Cs-134	No	3,500	Deselected	No	615	6.77/4.44	Yes	No	Distrib.	No	446	7.5%	No	Distribution
Cs-137	No	3,500	13.1 / 7.27	Very Low	615	14.18/7.75	Yes	No	Distrib.	No	446	Yes	No	Distribution
Eu-152	Positive	7,270	8.41 / 7.36	No	N/A	Deselected	No	No	Distrib.	Positive	148,000	<5%	No	Distribution
Eu-154	Positive	7,270	Deselected	No	N/A	Deselected	No	No	Distrib.	Positive	148,000	<5%	No	Distribution
Eu-155	Positive	7,270	Deselected	No	N/A	Deselected	No	No	Distrib.	Positive	148,000	<5%	No	Distribution
H-3	Negative	0.043	Deselected	No	N/A	Deselected	No	No	Distrib.	Negative	0.043	<5%	Negative	0.043
Ni-63	Positive	542	Deselected	No	62	3572/ 763	Yes	No	Distrib.	No	424	<5%	No	Distribution
Np-237	Negative	9.05	Deselected	No	N/A	Deselected	No	Positive	77.8	N/A	N/A	N/A	N/A	N/A
Sb-125	Positive	128	Deselected	No	N/A	Deselected	No	No	Distrib.	N/A	N/A	<5%	Positive	3,310
Pu-241	No	953	Deselected	No	N/A	Deselected	No	No	Distrib.	No	953	<5%	No	Distribution
Sr-90	Positive	168	Deselected	No	2.3	12.09/1.66	Yes	No	Distrib.	No	32	31%	No	Distribution
Tc-99	Positive	0.147	Deselected	No	N/A	Deselected	No	Positive	4.210	Positive	75.4	<5%	Positive	4.28

N/A – Radionuclide was not a radionuclide of concern and/or its dose sensitivity to Kd was not analyzed.

- Other radionuclides for which additional support in selecting a Kd value from a literature parameter range is unlikely considering the criteria stated in DWUP-ISG-002 (contingent on the results of characterization surveys):

  - C-14 - Radionuclide was detected in soil at three of the sites reviewed (CY, Rancho Seco and Yankee Rowe) at low fractions of the DCGL concentrations. Concerning the sensitivity of dose to the C-14 Kd, for the four approved LTPs reviewed, it was only sensitive for two cases (both positive correlations for Fort Calhoun and Connecticut Yankee). These experiences show that to use low Kds for C-14 would increase the unsaturated soil DCGL.

  - Fe-55 - Radionuclide has been detected but only at levels that are a small fraction of the very high Fe-55 DCGL (typical soil DCGL value 27,400 pCi/g for CY). We are not aware of any power plant site where Fe-55 has been detected in groundwater.

  - I-129 – Radionuclide has not been detected at the five plant sites reviewed and it was determined not to be a radionuclide of concern at any of these sites.

  - Tc-99 - Radionuclide has not been detected in significant concentrations nor has it been determined to be a radionuclide of concern at any of the 5 plants analyzed. It should be noted that concerning the sensitivity of dose to the Tc-99 Kd, for the four LTPs where sensitivity to Kd was analyzed, positive sensitivity was shown in all cases. These experiences show that using very low Kds for Tc-99 could increase the unsaturated soil DCGL.

  - Plutonium Radionucides were not detected in soil at concentrations that are significant compared to their DCGLs at the sites reviewed. Plutonium has not been detected in groundwater at any US power plant site.

  - Ni-63 - Radionuclide has been detected in low concentrations in soil and groundwater at a few plant sites. Ni-63 was deselected for Fort Calhoon as there were no detections there. Remediations at CY were not driven by Ni-63 as the concentrations were a very small fraction of the soil DCGLs (723 pCi/g).

    Concerning the sensitivity of dose to the Ni-63 Kd, for the four LTPs where sensitivity to Kd was analyzed, it was not sensitive for three cases and showed positive sensitivity in one case (Fort Calhoun). The lowest Kd used at any of these sites was 62 (from soil testing at Zion). Using higher Kds than this value would likely have less than a 1 percent increase of the unsaturated soil DCGL.

    Although most sites should be able to classify Ni-63 as insignificant, sites that detect high levels of Ni-63 in soil and/or groundwater (i.e., approaching the EPA MCLs) should consider further evaluation of the mobility of Ni-63.

  - H-3 - Radionuclide has been detected in soil and groundwater at many power plant sites (although mostly at relatively low concentrations). Tritium was deselected for Fort Calhoon, Rancho Seco and Zion as there was little or no detections there. Although elevated levels of H-3 were measured in groundwater at CY and Yankee Rowe, remediations at these sites were not driven by H-3, as the soil concentrations were a very small fraction of the DCGLs (lowest value, 372 pCi/g at Yankee Rowe).

Concerning the sensitivity of dose to the H-3 Kd, for the four LTPs where H-3 Kd sensitivity was analyzed, Kd was not a sensitive parameter for Rancho Seco and a negative sensitivity was shown in the other three cases. For these later three sites, a H-3 Kd of 0.043 was selected from the parameter range. DWUP-ISG-002 recommends a Kd of zero for H-3. The unsaturated soil DCGL analysis at CY showed that only 15.4 % of the dose from H-3 was from water dependent pathways. Use of a H-3 Kd value of zero could result in higher H-3 DCGLs for soil as the water independent pathways dominant the dose from unsaturated soil.

Although many sites may be able to classify H-3 as insignificant, sites that detect high levels of H-3 in soil and/or groundwater (i.e., approaching the EPA MCLs) will likely need to consider H-3 as a radionuclide of concern.

- Sr-90 – Although not a radionuclide for which DWUP-ISG-002 recommends further justification for the Kd selection, a few sites may consider it a radionuclide of concern if it is detected in high concentrations in soil and/or groundwater (i.e., approaching the EPA MCLs). At Connecticut Yankee, although shallow soil remediations were driven by Co-60 and Cs-137, deep soil and bedrock remediation was driven by Sr-90 to reduce levels of Sr-90 in groundwater below the EPA MCLs. Sites that have a similar situation to CY will likely have performed more extensive groundwater and soil characteristic studies such that information to support a selection of Kd for Sr-90 should be available.

  It is noteworthy that although the Kd value for Sr-90 determined by testing of soil at Zion was 2.3, the resulting soil DCGL calculated for Zion showed good agreement with the Sr-90 soil DCGLs calculated for Connecticut Yankee and Yankee Rowe where much higher Kds were used. One possible explanation is, for the four sites where the sensitivity of dose to the Sr-90 Kd was reviewed, no sensitivity was shown in three cases, and a positive sensitivity was shown in one cases (Fort Calhoun).

- Co-60 and Cs-137 – Although these are the primary radionuclides in contamination at power plant sites being decommissioned, dose has not been shown to sensitive to Kd for all sites reviewed except for a positive sensitivity for Co-60 at Fort Calhoun. Neither radionuclide has been detected in groundwater except for Cs-137 in wells very near two source areas at Connecticut Yankee. As the lowest Cs-137 Kd used at sites reviewed is 446 at CY, increasing Kd above this value would have had minimum effect on the Cs-137 DCGL.

Summary Effect of Kd on Unsaturated Soil DCGL

- At most nuclear power plant sites for which LTPs have been approved and the resident farmer scenario has been used in determining unsaturated soil DCGLs:
  - Only a few of radionuclides have been detected in concentrations that are greater than a few percent of the soil DCGLs and would therefore meet the criteria in DWUP-ISG-002 for further evaluation of the Kd parameter for use in dose modeling.
  - Of these few radionuclides, only H-3 has shown a sensitivity where dose could be significantly affected by the Kd selected.

- Power plant sites should review their soil and groundwater characterization results to determine if either of the two criteria stated in DWUP-ISG-002 apply and the need for further justification of the Kd parameter evaluated.

K.2 Effect of Kd on Saturated Soil DCGLs

For saturated soil at power plant sites dose is primarily due to the water dependent pathways. These sites also typically determine compliance dose by applying groundwater DCGLs or PDCF as discussed in Section F.4.2.

The Fort Calhoun site performed an uncertainty analysis for saturated soil which is summarized in the Fort Calhoun LTP (Reference 6-10). Table F-6-6 of this LTP shows that the results of that uncertainty analysis are that only Np-237 was sensitivity dose. As Np-237 has not been detected in significant quantities at power plant sites, this experience indicates that median Kd values from literature likely can be used for determination of saturated soil DCGLs for the significant radionuclides.

K.3 Effect of Kd on SSDM Wall/Floor DCGLs

Subsurface Structure Dose Models (SSDMs) for power plant sites typically assume that the radionuclides contained in the floors and walls of a building diffuse out of the concrete into the material used to fill the basement after it has been emptied of SSCs and backfilled. As the dose determined in these models is typically from water dependent pathways, the radioactivity is assumed to be released into saturated conditions. Although the rate of this diffusion used in this calculation can be based on literature values, there is also the option to make the conservative assumption that all that activity is instantaneously released into and mixed with the backfill.

The Fort Calhoun site performed an uncertainty analysis for the scenario discussed in the last paragraph assuming instantaneous release. Table F-14 shows the results of that uncertainty analysis. For all the radionuclides except Ce-144 and Fe-55 (for which kd was not sensitive to dose), there was a negative correlation of Kd to dose. This is consistent with the predominance of water dependent pathways with this scenario as lower Kds in a saturated zone will result in more radioactivity in groundwater. Fort Calhoun selected the 25th percentile Kd for sand for the sensitive radionuclides which was conservative (adding to the conservatism of the instantaneous release assumption) as the actual fill material was to be silt loam soil type which typically has higher Kds.

K.4 Effect of Kd on Buried Pipe DCGLs

Fort Calhoun

One of the scenarios used to determine the DCGL for buried pipe at Fort Calhoun was to conservatively assume that the pipe:

- Stays in place (i.e., in-situ);
- Is located under the water table in the saturated zone and
- Instantly dissolves at time=0 with its radioactivity content instantly mixed with the surrounding soil

A sensitivity analysis was performed for the physical parameters using this scenario. Table F-26 shows the results of this sensitivity analysis for Kd. Although Kd was shown to be negatively correlated to dose for a number of radionuclides, only C-14 was a radionuclide of concern (ROC) at Fort Calhoun. Although characterization results for Fort Calhoun showed significant levels of C-14 only in concrete, it was considered to be a ROC for all media.

Zion (TSD 14-015)

The Zion site also performed sensitivity analyses for various buried pipe scenarios using RESRAD. The three scenarios analyzed were:

- Excavation of the buried pipe
- Buried pipe left In-situ in the unsaturated zone
- Buried pipe left In-situ in the saturated zone

The results of these analyses were that only Sr-90 was sensitive to dose (negative correlation) and only for the saturated zone case. This is consistent with other experiences as for a saturated zone scenario, lower Kd would result in higher groundwater concentrations and higher dose from the water dependent pathways. For conservatism, Zion set all the Kd parameters used to determine the buried pipe DCGLs to the minimum site-specific value to maximize the groundwater concentrations.

Considering these experiences, additional justification for the use of literature values for Kd may not be required for buried pipe scenarios depending on the results of sensitivity analyses.

K.5 Effect of Kd on Excavation and Drilling Spoils DCGLs

The scenario for excavation and drilling spoils at Fort Calhoun assumes that the material is brought to the surface and spread on the soil there. As this scenario is the same as that for surface and unsaturated subsurface soil, the sensitivity analysis and the resulting Kds used to determine the soil DCGLs were also used to determine the excavation and drilling spoils DCGLs. As discussed above, there was no need for additional justification for the use of literature Kd values at Fort Calhoun as none of the significant radionuclides showed a sensitivity to dose that would decrease the unsaturated soil DCGLs.

K.6 References for Appendix K

K-1 *Haddam Neck Plant License Termination Plan*, Revision 1, Chapter 6.

K-2 ZionSolutions Technical Support Document 14-010, *RESRAD Dose Modeling for Basement Fill Model and Soil DCGL and Calculation of Basement Fill Model Dose Factors and DCGLs*, Revision 6, July 2017.

K-3 *Rancho Seco License Termination Plan*, Revision 1, May 2008, Chapter 6.

K-4 Yankee Atomic Calculation, YA-CALC-01-002-03, *Derived Concentration Guideline Levels at the Yankee Rowe Site*, 1/19/2004.

K-5 *Yankee Atomic Final Status Survey Report*, YNPS-FSS-NOL01-00, 3/29/06.