Guide to Assessing Radiological Elements for License Termination of Nuclear Power Plants

Technical Report

Guide to Assessing Radiological Elements for License Termination of Nuclear Power Plants

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Final Report, June 2002

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REPORT SUMMARY

This report provides guidance in the preparation of a License Termination Plan (LTP) to utilities engaged in nuclear plant decommissioning. The Nuclear Regulatory Commission requires utilities to submit the LTP document years prior to the site license termination. This report focuses on the radiological components of the LTP. It identifies and addresses the regulatory requirements of each element in a way useful to the utility end user.

Background

The new license termination regulations involve numerous complex regulatory guidance documents. This complexity has resulted in the majority of initial LTP submittals experiencing numerous setbacks or delays in the NRC approval process. This report is directed to assist those engaged in future preparation by offering a comprehensive, "User Friendly" guide that addresses the radiological components of license termination. This guide captures the lessons learned by those utilities currently undergoing the license termination process. This experience, coupled with consolidation of the important aspects of available regulatory guidance, will greatly benefit the next generation of utilities considering decommissioning and termination of their NRC Part 50 licenses.

Objective

To provide a guidance document to assist nuclear power station personnel in addressing the radiological aspects of license termination.

Approach

The project team reviewed applicable regulations, regulatory guidance documents, LTPs with their associated Requests for Additional Information (RIAs), and information received from decommissioning utilities. They then compiled the relevant information required to address the radiological aspects of license termination. A cross section of industry experts involved with the license termination process peer reviewed the resulting report.

Results

EPRI has developed a guide to assist nuclear power plant personnel in addressing the radiological aspects of license termination. The guide specifically addresses regulatory requirements and guidance, site characterization, dose modeling, site remediation, and final status survey. The appendices provide examples of a sampling plan and a conceptual schedule for site characterization. This report serves as a technical reference addressing a wide range of issues pertaining to the radiological aspects of license termination. The guidance provided in this report expands beyond the NRC requirements for submittal of a LTP.

EPRI Perspective

The Nuclear Regulatory Commission is currently reviewing and consolidating their regulatory guidance for Nuclear Power Plant License Termination based on licensee and regulator experiences gained during implementation of these relatively new NRC guidelines. This report offers comprehensive guidance related to radiological components of the LTP document. This EPRI report is one of a series published in 2002 addressing the central issue of nuclear plant license termination and site release. The other reports include:

Summary of License Termination Plans Submitted by Three Nuclear Power Plants (TR-1003426).

Use of Probabilistic Methods in Nuclear Power Plant Decommissioning Analysis Dose (TR-1006959).

Trojan Nuclear Plant License Termination Plan Development Project (TR-1003423).

EPRI will revise these guidance documents to reflect any significant modifications to license termination regulation, associated regulatory guides, or their regulatory interpretation as issued by the NRC.

Key Words

Decommissioning
License termination

ABSTRACT

This guide will assist nuclear power station personnel in addressing the radiological aspects of terminating their NRC Part 50 license. The guide specifically addresses the following key areas: regulatory requirements and guidance, site characterization, dose modeling and compliance with radiological criteria, site remediation, and final status survey. The appendices provide examples of a sampling plan and a conceptual schedule for site characterization.

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GLOSSARY

Acronyms and Abbreviations

ALARA	As Low As is Reasonably Achievable
ANSI	American National Standards Institute
ASTM	American Society for Testing and Materials
BWR	Boiling Water Reactor
CFR	Code of Federal Regulations
DandD	(Decontamination and Decommissioning) NRC computer dose modeling code
DCGL	Derived Concentration Guideline Level
$DCGL_{EMC}$	Derived Concentration Guideline Level used for the Elevated Measurement Comparison
$DCGL_{eff}$	Effective Derived Concentration Guideline Level
$DCGL_{n}$	Nuclide specific Effective Derived Concentration Guideline Level
$DCGL_{W}$	Derived Concentration Guideline Level used for the statistical tests
DF	Decayed Fraction
DOE	Department of Energy
DQA	Data Quality Assessment
DQO	Data Quality Objective
EPA	U.S. Environmental Protection Agency
EPRI	Electric Power Research Institute

ER	Environmental Report
FSAR	Final Safety Analysis Report
FSS	Final Status Survey
GLV	Guideline Value
GPS	Global Positioning System
H_0	Null Hypothesis
HSA	Historical Site Assessment
HTDR	Hard To Detect Radionuclide
ICRP	International Commission on Radiological Protection
LBGR	Lower Bound of the Gray Region
LHS	Latin Hypercube Sampling
LIMS	Laboratory Information Management System
LTP	License Termination Plan
MARSSIM	Multi-Agency Radiation Survey and Site Investigation Manual
MDC	Minimum Detectable Concentration
MDL	Minimum Detectable Limit
NMSS	Nuclear Material Safety and Safeguards
NPDES	National Pollutant Discharge Elimination System
NPP	Nuclear Power Plant
NRC	U.S. Nuclear Regulatory Commission
ORISE	Oak Ridge Institute for Science and Education
PC	Protective Clothing
PCB	Polychlorinated Biphenyls
PE	Performance Evaluation

PWR	Pressurized Water Reactor
QA	Quality Assurance
QAPP	Quality Assurance Project Plan
QC	Quality Control
RCRA	Resource Conservation and Recovery Act
REMP	Radiological Environmental Monitoring Program
RESRAD	(Residual Radioactivity) Computer dose modeling code for soil contaminated with radionuclides
RESRAD-BUILD	Computer dose modeling code for buildings contaminated with radionuclides
RIS	NRC Regulatory Issue Summary
RP	Radiation Protection
RWP	Radiological Work Permit
SOP	Standard Operating Protocol
TEDE	Total Effective Dose Equivalent
TPH	Total Petroleum Hydrocarbon
USCS	Unified Soil Classification System
V&V	Verification and Validation
WP	Water Pollution
WRS	Wilcoxon Rank Sum
WS	Water Supply

UNITS OF MEASUREMENT

cm	centimeter
cpm	counts per minute
dpm	disintegrations per minute
dpm/100cm^2	disintegrations per minute per 100 square centimeters
ft.	foot
gallon	unit of liquid volume
in.	inch
L	liter
m^2	square meter
m^3	cubic meter
MeV	mega electron volts
mrem	millirem
mSv	millisievert
µCi	microcurie
pCi	picocurie
pCi/g	picocurie per gram (unit of radionuclide concentration)
psi	pound per square inch
yr	year

SI Unit Conversion

Unit Name	Unit Symbol	Converted to SI Units
disintegrations per minute	dpm	0.0167 Bq
disintegrations per minute per 100 square centimeters	dpm/100 cm^2	0.0167 Bq/100cm^2
foot	ft.	30.48 cm 0.3048 m
gallon	gallon	0.004 m^3
square foot	ft^2	929 cm^2 0.0929 m^2
inch	in.	2.54 cm 0.0254 m
liter	L	1,000 cm^3 0.001 m^3
millirem	mrem	0.01 mSv
microcuries	μCi	37,000 Bq
picocurie	pCi	0.037 Bq
picocurie per gram	pCi/g	37 Bq/kg
pounds per square inch	psi	7,000 Pa

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1 INTRODUCTION

This EPRI document provides guidance for addressing the radiological elements of the license termination process, specifically, site characterization, dose modeling, site remediation, and final status survey. It should be noted that this guidance expands beyond what is required by NRC regulation for submittal of a License Termination Plan.

The requirements for a License Termination Plan (LTP) for a nuclear power plant are stipulated in 10 CFR 50.82, "Termination of License". This plan must be submitted to the NRC at least 2 years prior to the license termination date. Under this regulation, the following areas are required to be discussed:

"The license termination plan must include –

(A) A site characterization;

(B) Identification of remaining dismantlement activities;

(C) Plans for site remediation;

(D) Detailed plans for the final radiation survey;

(E) A description of the end use of the site, if restricted;

(F) An updated site-specific estimate of remaining decommissioning costs; and

(G) A supplement to the environmental report, pursuant to §51.53, describing any new information or significant environmental change associated with the licensee's proposed termination activities."

The site characterization section of this report describes the methods used for determination of the identity and magnitude of radioactive materials in all relevant media at a site. This involves the development of information that feeds into final status survey planning, dose assessment, and remediation actions.

The dose modeling section discusses the radiological criteria for site release and the critical population group concept for performing dose analyses. The available dose modeling computer codes are described and the use of these codes in performing screening and site-specific analyses is discussed. The final section summarizes work that has been performed with the RESRAD code in the probabilistic mode. The results are presented as probabilistic distributions and these results are compared to published screening DCGLs.

The site remediation section discusses various techniques that may be used to decontaminate soils, buildings, and systems. This section also addresses ALARA evaluations of remediation efforts, as required by NRC. In addition, some industry cost data that was used in actual site ALARA evaluations are provided.

The Final Status Survey (FSS) section provides a simplified overview of the FSS process, along with a description of the basic elements in the Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM) (Reference 1) that are common to most site FSS plans. The section directs the user back to the applicable sections of MARSSIM guidance where more detail can be obtained.

1.1 Regulatory Framework

The LTP describes, in detail, how a licensee will comply with the NRC's requirements for site release. When developing such a plan, it is important to understand what is required by regulation, and to review the available guidance to assist in interpretation and compliance.

Several licensees have submitted License Termination Plans for NRC review. These reviews have identified common areas of deficiency that have resulted in the NRC issuing requests for additional information (RAIs) to the licensees (Reference 2) [These areas of deficiency are further discussed in Section 1.2]. By clearly understanding the license termination process, the applicable regulations, and subsequent revisions, the licensee may be able to reduce both the number of RAIs, and additional costs and delays that may be incurred. Cited below are the license termination requirements that a site must comply with, followed by a brief discussion of those documents intended to provide guidance on acceptable methods, delineation of useful techniques, and clarification of requirements through examples.

1.1.1 Regulatory Requirements

The process for terminating a radioactive material license is outlined in the Code of Federal Regulations (CFR). Specific regulations that apply to terminating a nuclear power reactor license are indicated below.

1) 10 CFR 20, Subpart E, Radiological Criteria for License Termination

Under 10 CFR 20, Subpart E, regulations are cited which apply to the decommissioning of sites which are licensed under 10 CFR 30, 40, 50, 60, 61, 70, and 72, in addition to other sites which are subject to The Nuclear Regulatory Commission's jurisdiction. These regulations set up release criteria, which must be adhered to by nuclear power plants undergoing license termination.

A critical subpart E citation for license termination of a nuclear power plant, intended for unrestricted release, is excerpted below from 10 CFR 20.1402:

> *"A site will be considered acceptable for unrestricted use if the residual radioactivity that is distinguishable from background radiation results in a TEDE to an average member of the critical group that does not exceed 25 mrem (0.25 mSv) per year, including that from groundwater sources of drinking water, and the residual radioactivity has been reduced to levels that are as low as reasonably achievable (ALARA). Determination of the levels which are ALARA must take into account*

consideration of any detriments, such as deaths from transportation accidents, expected to potentially result from decontamination and waste disposal."

2) 10 CFR 50.75, Reporting and Record Keeping for Decommissioning Planning

10 CFR 50.75 outlines the requirements for records of site environmental events relevant to the decommissioning of a nuclear power plant site. Specifically, reports and records for the above purpose must be generated and maintained in a site file, as stated in the following citation:

"(g) Each licensee shall keep records of information important to the safe and effective decommissioning of the facility in an identified location until the license is terminated by the Commission. If records of relevant information are kept for other purposes, reference to these records and their locations may be used. Information the Commission considers important to decommissioning consists of –

(1) Records of spills or other unusual occurrences involving the spread of contamination in and around the facility, equipment, or site. These records may be limited to instances when significant contamination remains after any cleanup procedures or when there is reasonable likelihood that contaminants may have spread to inaccessible areas as in the case of possible seepage into porous materials such as concrete. These records must include any known information on identification of involved nuclides, quantities, forms, and concentrations.

(2) As-built drawings and modifications of structures and equipment in restricted areas where radioactive materials are used and/or stored and of locations of possible inaccessible contamination such as buried pipes, which may be subject to contamination. If required drawings are referenced, each relevant document need not be indexed individually. If drawings are not available, the licensee shall substitute appropriate records of available information concerning these areas and locations.

(3) Records of the cost estimate performed for the decommissioning funding plan or of the amount certified for decommissioning, and records of the funding method used for assuring funds if either a funding plan or certification is used."

3) 10 CFR 50.82, Termination of License

This section of the regulations addresses termination of both power and non-power reactor licenses. Section 50.82(a)(9) of this regulation discusses preparation of the LTP for a nuclear power reactor licensee. The specific elements of the LTP have already been mentioned above. In addition, 10 CFR 50.82(a)(9) states the following:

"All power reactor licensees must submit an application for termination of license. The application for termination of license must be accompanied or preceded by a license termination plan to be submitted for NRC approval.

(i) The license termination plan must be a supplement to the FSAR or equivalent and must be submitted at least 2 years before termination of the license date."

1.1.2 Implementation Guides

The NRC has published a series of guidance documents pertaining to license termination that are comprised mainly of Regulatory Guides and NUREGs. Users of these documents should also be cognizant of current applicable Information Notices. The intent of this regulatory guidance is to provide a more detailed description of a process the licensee can follow in order to demonstrate compliance with the regulations. These guides describe methods and techniques employed by the NRC to both analyze specific events and review applications for licenses and subsequent amendments. Specific to such guides are standard formats, which are used by the NRC staff in the review of submitted documents, such as a LTP and a Decommissioning Plan (DP).

The three main objectives of the LTP are 1) to demonstrate that there are adequate funds to complete decommissioning and release the site for unrestricted (or restricted) use, 2) to demonstrate that the site release criteria ensure that exposure levels are kept as low as reasonably achievable (ALARA), and 3) to demonstrate that the final status survey program is adequate to release the site for unrestricted (or restricted) use.

The primary regulatory guide pertaining to license termination is RG 1.179, Standard Format and Content of License Termination Plans for Nuclear Power Reactors (typically used with NUREG 1700 (Reference 3)). This guide represents a good starting point for licensees, given that an outline of the LTP content and regulatory basis is provided. The licensee can then begin to compile the additional guidance that will be necessary to complete the LTP.

The true “guidance” documents that provide the information necessary to address each section of the LTP are the NRC NUREGs, which are comprised of a series of reports, manuals, procedural guidance, newsletters, papers, and books of a technical nature. Although this list is not complete, the following NUREGs contain a substantial amount of guidance, which include descriptions of the planning, development, implementation, and evaluation of a standard LTP, and are the guidance documents that are most often cited in site LTPs.

1) NUREG-1700, Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans.

2) NUREG-1727, NMSS Decommissioning Standard Review Plan.

The information presented in NUREG-1727 is intended to supplement that provided by NUREG-1700 in the areas of site characterization, dose modeling, final radiation surveys, and institutional controls. Since this NUREG outlines the necessary guidance in a generic format, the licensee should review the entire document for those details that pertain to the site (Reference 2).

The Final Status Survey (FSS) Plan is usually incorporated into the LTP. The method for performing the FSS has undergone major revisions in the past several years. The following documents are references for the development of a site-specific FSS plan.

1) NUREG-1505, "A Nonparametric Statistical Methodology for the Design and Analysis of the Final Status Decommissioning Survey."

2) NUREG-1507, “Minimum Detectable Concentrations with Typical Radiation Survey Instruments for Various Contaminants and Field Conditions.”

3) DRAFT NUREG-1549, “Decision Methods for Dose Assessment to Comply with Radiological Criteria for License Termination.”

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

5) NUREG/CR-5512, “Residual Radioactive Contamination from Decommissioning,” Volumes 1 - 3.

6) NUREG/CR-5849, “Manual for Conducting Radiological Surveys in Support of License Termination.”

7) NUREG/CR-6692, “Probabilistic Modules for the RESRAD and RESRAD-Build Computer Codes.”

1.2 Lessons Learned

Currently, guidance on license termination is relatively new and is still undergoing revision. LTPs that have been submitted have undergone several review iterations prior to receiving final approval. Therefore, it is important to understand the lessons learned in order to simplify the review and approval process.

In October 2000, the Decommissioning Branch Chief of NRC’s Division of Waste Management summarized LTP submittals to date and the resulting approvals. For LTPs submitted under the new guidance, only one of three had received approval at the time of the presentation (Reference 4). The approval of the one LTP was attributed to two factors: 1) the content was consistent with NUREG-1700, and 2) the licensee had incorporated the lessons learned from previous sites.

Similarly, a “lessons learned” document was issued in January of 2002 as an NRC Regulatory Issue Summary (RIS). RIS 2002-02, Lessons Learned Related to Recently Submitted Decommissioning Plans and License Termination Plans, identified several common areas of review that resulted in requests for additional information (RAIs) from licensees submitting Decommissioning Plans or License Termination Plans. The areas identified as commonly falling short of the necessary detail include communications between NRC staff and the licensee during preparation of the LTP, data quality objectives, understanding of flexibility in the applicable NUREGs (e.g., NUREG-1575 and NUREG-1727), dose modeling issues, use of records in the historical site assessment/site characterization, and classification of survey units. Having identified these areas, the NRC staff has responded by adding an additional step to the review process. The additional step comes in the form of a limited technical review to identify significant technical deficiencies before an LTP is submitted for a detailed review. The limited technical review is generally narrowed to specific areas where past experience has typically denoted a need for more detail. The LTP components that have been listed are site characterization, dose modeling, the final radiation survey, the decommissioning cost estimate, and institutional controls (applicable only to restricted release) (Reference 2).

As stated above, a useful tool in the planning and development of a License Termination Plan is effective communication with the NRC staff. The following RIS citations discuss such consultations:

> *"Early and frequent consultations between U.S. Nuclear Regulatory Commission (NRC) staff and licensees are encouraged during the planning and scoping phase supporting the preparation of … LTPs. In this context, a licensee may schedule a meeting with the NRC Project Manager assigned to the site to discuss the planning and content of the LTP… The discussions would address, 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; decommissioning goals (restricted vs. unrestricted license termination); basis for cleanup criteria and development of site-specific derived concentration guideline levels (DCGLs), or commitment to use NRC default DCGLs; potential impact on public health and safety or the environment; funding plan and financial assurance; and the minimum information required to be contained in the LTP…"*

In addition to this report, there exists a parallel EPRI guidance document, The License Termination Plan Summary Report (projected publication date is July 2002), which summarizes the results of LTPs submitted to the NRC by Connecticut Yankee, Maine Yankee, and the Trojan Nuclear Plant.

1.3 Consolidation of Guidance Documents

The NRC has identified the need to consolidate approximately 80 decommissioning guidance documents into a central resource. This resource will be a three-volume NUREG that will encompass regulatory guides and NUREGs, as well as technical assistance requests, decommissioning licensing conditions, and all generic decommissioning communications generated over the past few years. This NUREG is currently in the process of development and will include three functional topics: Volume 1) the decommissioning process (Reference 5), Volume 2) characterization, survey, and determination of radiological criteria, and Volume 3) financial assurance, record keeping, and timeliness. When this series is completed the resulting guidance will identify those areas of information necessary for the termination of a license and will supercede NUREG-1727, NMSS Decommissioning Standard Review Plan and NUREG/BR-0241, NMSS Handbook for Decommissioning Fuel Cycle and Materials Licensees.

Volume 1 of the series, Consolidated NMSS Decommissioning Guidance: Decommissioning Process, was issued as a draft report in January of 2002. The purpose of this NUREG is to provide a consolidated guidance document for licensees, applicants, NRC license reviewers, and other NRC staff (Reference 5).

2
SITE CHARACTERIZATION

As stated in the previous section, this report provides guidance that expands beyond what is required by NRC regulation for submittal of a License Termination Plan. Since site characterization is such a key component of terminating a license and continues throughout the decommissioning process, an in depth discussion of this subject is provided below.

Site characterization is defined as the determination of the extent and range of radioactivity in media at a site (Reference 6). This may include:

- Structures
- Systems
- Components
- Residues
- Soils and Sediment
- Surface and ground water

Characterization studies have three primary objectives:

- Ensure that the final survey efforts will provide the expected results,
- Develop the required input for a decommissioning plan and dose assessments, and
- Support the evaluation of remediation alternatives and technologies

Since radiological systems and components are generally removed from a site for appropriate disposal, data from their characterization is not likely to be a primary part of the characterization information supporting license termination. Likewise, extensive characterization data should not be necessary for buildings that will be removed prior to license termination.

While site characterization work is an integral part of decommissioning studies, it is not emphasized in great detail within the regulations. This is because the main objective of characterization is to provide confidence that a final survey can be performed with satisfactory (no unexpected) results. Thus, regulators carefully scrutinize final survey results, while the responsibility to judge the results of site characterization is largely placed on the licensee. Thus, characterization, while relatively flexible, can be challenging.

There is, however, a great deal of guidance and procedural information available to help with characterization programs. Most programs for decommissioning nuclear power plants (NPPs) will use MARSSIM (Reference 1) to plan for characterization and to accomplish final radiological surveys. This chapter contains the primary objectives necessary to provide a structure for characterization and to identify pertinent references.

Coordination between site characterization and final survey work will require that characterization methods focus on final survey requirements. The final survey will be carried out for all impacted areas of the site. The licensee will also use site characterization information for the development of dose modeling inputs, general purposes, and/or for the purpose of developing Derived Concentration Guideline Levels (DCGLs), which are essentially site release criteria. The decommissioning plan uses the results of Site Scoping (a preliminary characterization effort) to help assess the scope and planning of decommissioning activities. The LTP uses site characterization data in the same way, yet at a later stage of the process.

Site characterization will also help guide the true scope of site decommissioning work, when appropriately integrated into a project. Thus, proper attention to the details of site characterization can significantly improve a project's schedule and costs.

Good characterization programs begin with and rely on preliminary planning and proper project structuring. The flow chart in Figure 2-1, adapted from Reference 1, provides an overview and depiction of the elements of site characterization from the perspective of the final survey. The four shaded rectangles on the chart show the primary tasks of a characterization program:

- Historical Site Assessment
- Scoping
- Characterization
- Remediation Support

These provide the data needed to classify a site into areas of uniform contamination. There are, however, many detailed objectives required to attain these goals, which include preparation, implementation, and documentation.

Appendix B contains a conceptual schedule for site characterization. It includes all the major categories of site characterization work and identifies prerequisites and resources for the elements discussed later in this chapter.

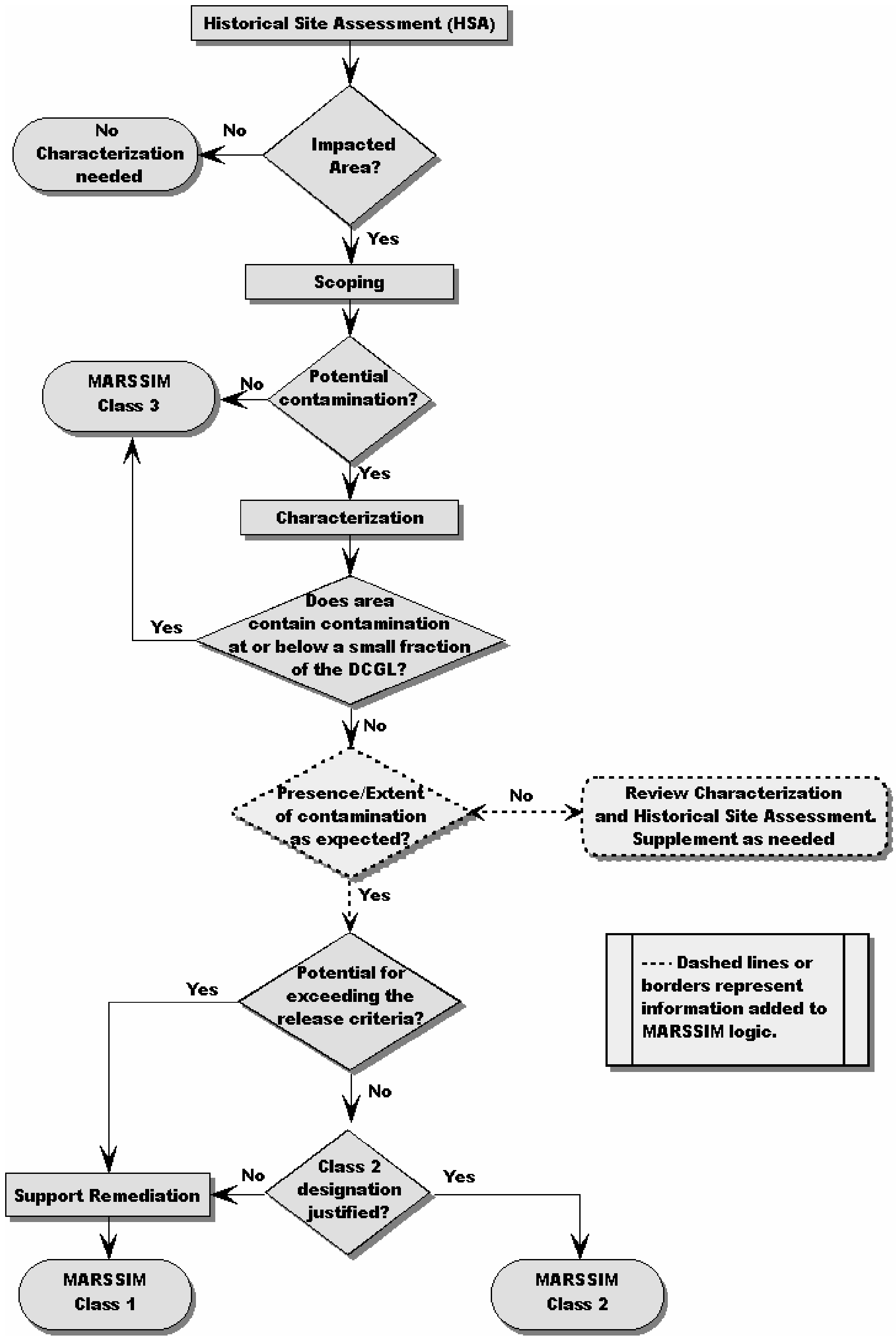

Figure 2-1
Site Characterization Work in MARSSIM Context

2.1 Project Initiation and Administration

Initial project planning requires that decisions be made regarding the overall project goals. Important factors to be considered would include:

- Site release conditions: unrestricted or restricted release
- Demolish buildings or leave in place
- Removal of building rubble or use of rubble as backfill
- Future site use and any necessary conditions
- Division of the site into areas with different release criteria
- Specific commitments to the regulators regarding site release criteria

The scope and initial planning of the characterization work will depend heavily on these goals. Only with clear information about these controlling issues can site characterization proceed efficiently. Their clear delineation will form a solid basis for all ensuing work.

A project team with experience and knowledge of the administrative and performance processes will help to initiate the program, minimizing the chance of significant hidden issues. Full review and documentation of all applicable regulations (i.e., federal, state, and local) should be conducted to ensure continuity of the project objectives.

An overall administrative procedure that governs all of the activities and puts all of the work procedures in perspective is also an important part of project initiation.

2.1.1 Project Team

The next step for a site characterization project is to assemble a capable team to plan, guide, and implement the necessary tasks. Resource requirements will include:

- Site characterization project management
- Planning and scheduling
- Field sampling
- Laboratory analysis
- Data documentation
- Data analysis
- Report writing
- Support staff (RP, etc.)
- Coordination with other decommissioning project tasks

A decommissioning nuclear power plant site may have a partial organization left in place from its operational staff. How the old and new staff integrates and interfaces should be clearly established and documented using organization charts and job descriptions. Responsibilities and support duties should be assigned for each element of the project, with priorities made clear to all those involved.

2.1.2 Regulations and Guidance

Environmental requirements related to the characterization of a nuclear power plant site are set out principally in the following regulations:

- 10 CFR 50.75, Reporting and Record Keeping for Decommissioning Planning.
- 10 CFR 50.82, Application for Termination of License.

10 CFR 50.75 requires that all site environmental events, relevant to decommissioning, be documented and maintained in a site file. 10 CFR 50.82 indicates that a LTP must include a site characterization section, but does not provide in-depth guidance on this requirement.

Regulatory Guide 1.179 (Reference 6) defines the purpose of site characterization for regulatory purposes as:

> *"…to ensure that the final radiation surveys are conducted to cover all areas where contamination existed, remains, or has the potential to exist or remain."*

Reference 7 also documents the need for a "site characterization package" separate from or part of the LTP submittal. Methods can be inferred from guidance chosen to address the final radiological survey, such as those found in Reference 8.

As discussed in Chapter 1, the licensee is required to submit a LTP within 2 years of the expected license termination. It must be submitted as a supplement to the licensee's Final Safety Analysis Report (FSAR) or as an equivalent document. The plan addresses each of the following tasks, pertinent to site investigations:

- Site characterization
- Identification of remaining site dismantlement activities
- Plans for site remediation
- Detailed plans for final radiation surveys for release of the site
- Methods for demonstrating compliance with the radiological criteria for license termination
- Supplement to the plant's Environmental Report if any new information or significant environmental changes are associated with the licensee's proposed termination activities

The NRC uses a standard review plan (Reference 8) to ensure quality and uniformity in its review process and to present a well-defined base from which to evaluate LTP submittals. The standard review plan also ensures that the NRC's review process is clearly understood.

To support radiological environmental investigations, guidance about a nuclear power plant site is available from a plant's Final Safety Analysis Report (FSAR) and Environmental Report (ER). A plant's Radiological Environmental Monitoring Program (REMP) (References 9 and 10) can also provide long term data on radiological results of analyses in soil, ground water, surface water and other various parameters from site and nearby off-site locations. Sampling requirements will vary from plant to plant based on the facility's Technical Specifications, which are derived from 10 CFR 50.

Among the best guidance documents available for the implementation of MARSSIM are the training manuals (References 11 and 12). These can provide significant information about specific site characterization elements and expand on a number of key issues that are not detailed in MARSSIM (Reference 1).

A database of regulations and guidance documents prepared and maintained as part of decommissioning licensing efforts would be a helpful resource for keeping site investigation work properly focused. Reference 11 contains a compilation of such documents.

2.1.3 Administrative Procedures

Two key administrative procedures prepared for a site characterization project will help to ensure efficient implementation and adequate results. These include: 1) a general administrative procedure to guide characterization and 2) a quality assurance program document that covers the characterization work to be done.

General administrative procedures provide information about:

- All other procedures used in characterization work,
- Potential media and analytes of concern,
- General project objectives,
- References to pertinent regulations, and
- Definitions of elements used in the planning phase.

Specific issue guidance, such as that needed to initiate sampling for a specific area could also be included. For example:

1) Describe the general process used for characterizing an area
    a) Definition of objectives and site boundaries
    b) Review/Assessment of available data and information about the site (including a historical site assessment)
    c) Development of a detailed sampling plan
    d) Collection of field samples or measurements in accordance with the sampling plan
    e) Laboratory analysis of samples

f) Evaluation of field and laboratory data

g) Re-sampling, as necessary, based on results of previous characterization

2) Describe the potential scope of existing data to be considered

a) Radiation Protection survey records and reports

b) REMP results

c) Scoping survey results

d) Interviews with employees and staff (former or current)

e) Site use and operations history

f) Plant drawings

g) Historic aerial photos

h) Geological and geophysical survey reports

i) NRC and state inspection reports

j) Corrective Action Program reports

3) Results of an area walk-down:

a) A check on the accuracy and completeness of existing information

b) Identification of areas needing additional information

c) Knowledge of security and site access restrictions

d) Any needs for immediate remedial action

e) Familiarity with topography and drainage

f) Possible delays in site characterization work

4) Sampling plan preparation

a) Purpose and scope of sampling

b) Summary of relevant historical data

c) Identification of known contaminants

d) Location of samples (using a map or measurements from fixed locations)

e) Methods for identifying sampling locations (e.g., GPS, hand measurements, stakes)

f) Documentation to ensure that the plan has been reviewed and approved prior to the initiation of sampling activities

Quality Assurance (QA) and Quality Control (QC) processes are an integral part of site characterization work. There are two general ways to implement a QA program for site characterization. The plant's existing 10 CFR 50, Appendix B program processes and procedures, adapted for high volume and detail of sampling, may be used. Alternatively, a Quality Assurance Project Plan (QAPP) can be prepared and properly adapted to sampling activities for the site. If characterization samples are to be used as final survey samples, then the same QA/QC standards should apply to both programs.

The basis of a QAPP is found in Reference 13, written by the U. S. EPA for the investigation of RCRA sites. MARSSIM (Reference 1) describes a QAPP in Appendix N. Among the useful concepts provided are data quality elements such as:

- Precision: measuring the reproducibility of measurements.
- Accuracy: statistical measurement of correctness including components of random error (variability due to imprecision) and systemic error; therefore, the total error associated with a measurement.
- Representativeness: achieved through use of the standard field, sampling, and analytical procedures, appropriate program design and consideration of elements, such as proper well locations.
- Completeness: calculated for the aggregation of data for each analyte measured for any particular sampling event or other defined set of samples
- Comparability: the confidence with which one data set can be compared to another data set. The objective for this QA/QC program is to produce data with sufficient comparability to make reliable project decisions.

Quality assessment for field samples may include:

- Ambient blanks used to assess the potential introduction of contaminants from ambient sampling conditions during sample collection.
- Equipment blanks used to assess the effectiveness of equipment decontamination procedures.
- Trip blanks used to assess the potential introduction of contaminants from sample containers or during transportation and storage prior to analysis.
- Field replicates used to assess precision of the sample collection process given two closely located samples.
- Field duplicates used to assess precision of the field sample processing operation.

Quality assessment for laboratory samples may include:

- Laboratory control samples used to evaluate the variability introduced during the preparation and analysis of each analytical batch and to determine if the method is in control.
- Matrix spikes and matrix spike duplicates used to document the bias of a method due to the sample matrix.
- Method blanks used to document contamination resulting from the analytical process.

As a result of field or laboratory quality assessments, data may be reported with the following type of qualifiers:

- The analyte was positively identified. The reported value is an estimate.
- The analyte was not detected. The result is at or below the Minimum Detectable Limit (MDL).
- The analyte was positively identified but the associated numerical value is below any release limit (DCGL).
- The data are unusable due to the inability to analyze the sample and meet the QC criteria.
- The analyte was found in an associated blank, as well as in the sample.
- A matrix effect was present.
- Sample for field screening data only.
- Analyte was identified without difficulty.

2.2 Project Planning

Project planning will require several elements to structure an effective site characterization project. Principal among these are: a multi-phase site history research program, establishment of Data Quality Objectives, and establishment of Analytes and Media of Concern.

2.2.1 Historical Site Assessment

A key starting point to organize a site characterization is the compilation of a Historical Site Assessment (HSA). MARSSIM, Section 3 provides a comprehensive description of this task.

2.2.1.1 Documentation survey

The objectives of an HSA include:

- The identification of known or potential sources of contamination,
- A determination of whether the site, inside or outside the facility, poses a potential threat to human health and the environment,
- Differentiation between impacted and non-impacted areas,
- Input for scoping and characterization surveys,
- An assessment of the likelihood of contaminant migration, especially off-site migration, and
- Identification of any neighboring site emissions that could interfere with data collected.

Historical assessment approaches can be made by describing history by areas, analytes, episodes, or by some combination of the three.

For older facilities, or for those that might not possess a fully documented history, a historical document search, which achieves a result "as thorough as reasonably possible", is typically performed. As a consequence, sampling may be more likely to produce unanticipated results. Such outcomes should prompt a re-examination of the site history to support the results, or prompt the recall of a historical source overlooked by an initial inquiry.

Figure 2-1 demonstrates that site history may not provide a basis for all substantial contamination found at a site. Given unexpected results, a second look at site history may help to provide a basis to support the outcome. It may provide confidence that results represent circumstances that might have been anticipated and not an unexplained occurrence of contamination. Amendments to the HSA, based on such findings, provide both valuable support and a reasonable basis for such investigations.

A complete review of site files, guided by knowledgeable site personnel, will provide an effective approach to this task. Valuable information on constructing an HSA will be contained in documents that address compliance with:

- 10 CFR 20.302 (Reference 14) or 10 CFR 20.2002 (Reference 15) regulations that address compliance with requirements for on-site disposal of waste, and
- 10CFR50.75g (Reference 16), which addresses residual contamination from site cleanups, documented specifically for decommissioning (Note that in many older facilities, 10 CFR 50.75 documentation may be incomplete, unreliable, or non-existent).

Other documents to be reviewed are:

- Routine radiological surveys done as part of plant operation
- REMP reports and raw data
- Site walk-down results
- Interviews with plant employees (current or former)
- Plant drawings and specifications
- Historical aerial photograph compilation and interpretation
- Geological and geophysical reports
- NRC and state inspection reports
- Permit/license history
- Radiological material purchase and disposal records
- Hazardous Material Inventory Records and Reports
- Corrective Action Program reports

In many cases original historic documents reviewed for site history will be referred to repeatedly over the course of a site investigation. For this reason, a database of documents assessed for the HSA can be a very useful tool. Such a database might include document references, sources and locations, and keywords relevant to the investigation such as site locations, potential contaminants, and incidents.

The licensee should also be cognizant that site records, as a singular source, used in the development of the HSA, may not provide a sufficient level of information. NRC experience indicates that submitted LTPs have sometimes contained these records along with operational and post-shutdown surveys as replacements for site characterization survey data. These substitutions could be considered inadequate as they may contain information that is only relevant for a limited time span or specific set of conditions. To fill in such possible gaps, additional information could be collected, such as through personnel interviews with staff (both present and past) and contractors (see Section 2.2.1.3). Thus, the NRC suggests a continuing evaluation of the HSA in conjunction with planning and/or execution of site characterization (Reference 2). Note that Figure 2-1 also addresses this iterative process in the review of the HSA and site characterization within the MARSSIM context.

2.2.1.2 Non-Radiological History of a Site

In accordance with the National Environmental Protection Act (NEPA) and the NRC's approval process for a LTP, an assessment must be made regarding the impacts on the environment from both radiological and non-radiological sources. It is the experience of the NRC that most licensees demonstrate effective radiological evaluations, while some show deficiencies in the non-radiological assessment (Reference 2).

In terms of the non-radiological investigation of site history, ASTM Standard E1527-94 (Reference 17) can serve as a suitable basis. Results from investigations based on this standard are typically known as "Phase 1 Investigations". Preparation of such an investigation requires:

- Records review
- Site reconnaissance
- Interviews
- Evaluation and reporting

Issues addressed include utilities, such as use of:

- Fuel oil heating systems
- Potable well water use
- Septic system use
- Incinerator use

Also addressed would be:

- Historical use and storage of oil and hazardous materials
- RCRA status and any RCRA compliance reviews
- Permit reviews including:

  - NPDES
  - Reclamation-type permits
  - Air permits
  - Solid waste permits
- Spill reporting and contingency planning review
- Solid waste disposal practices review
- Underground and above-ground storage tank history, inventory, and evaluations
- Waste water treatment
- Other historic industrial uses of the site, such as dry-cleaning for PCs
- PCB use (e.g., in transformer oil and paint)
- Herbicide/pesticide use (for transmission line maintenance)
- Lead in soil at any "practice" shooting range used by site security
- Use or disposal of soil excavated from the site for construction or other purposes during the time period of plant operation.

The investigation of a site for non-radiological contaminants is a critical step in site characterization. Investigation for radionuclides performed in conjunction with non-radiological investigations can optimize characterization efficiency and costs especially in areas where access may be limited, such as scaffold construction, confined space access, and sample retrieval.

2.2.1.3 Personnel Interview Survey

Personnel interviews should be comprehensive and allow for as much input as possible. Results of this investigation will collect accurate data and information that may not be verifiable in the absence of field investigation. The timing of interviews, conducted after the point at which documented historical data have been assembled and studied by interviewers, can enhance results by allowing topics to be mutually understood.

Interviewees are sometimes reluctant to provide information on historic activities due to fear of reprisal for themselves or others. The fact that there will be no reprisals should be clearly stated prior to interviews.

2.2.2 Data Quality Objectives and Planning

The Data Quality Objective (DQO) process, as described in MARSSIM (Reference 1), Appendix D, entails significant planning, assessment, implementation, and evaluation for a project prior to or at the beginning of the work. The DQO process will provide primary input to a Quality Assurance Project Plan (QAPP, Reference 1, Section 4.9), defining in detail how all quality measures are implemented for the project.

DQOs ensure that the survey results are of sufficient quality and quantity to support the license termination. The DQOs systematically define criteria for data collection. An example of general DQO planning for site characterization might include the following:

1) State the problem or objective:
   a) Determine the radiological condition of the site and establish the extent and degree of radioactive contamination for all site areas. Justify area classifications according to MARSSIM.
2) Identify the decision:
   a) Identify decisions and alternative actions.
   b) Identify the most appropriate scenario for the site.
   c) Areas, as defined, will either pass or fail a concentration test for analytes of concern in a given medium.
3) Identify the decision data requirements:
   a) Determine the concentrations of radionuclides in all media, including the distribution and history of radionuclides for a given area.
   b) Identify pertinent information sources.
   c) Identify the analytes of concern for the site.
4) Establish geographic, temporal and analytical boundaries:
   a) Provide maps with MARSSIM area boundaries identified.
   b) Establish decision-making deadlines.
   c) Describe surface/subsurface conditions, or seasonal variations that might influence analytical results.
   d) Establish analytical requirements for minimum detectable concentrations (MDCs) for analyses.
5) Develop decision rules: logical “if…then” statements defining potential conditions and the resulting actions:
   a) If unexpectedly high levels of radioactivity or “hot-spots” are detected in samples from a given area, then appropriate measures for re-sampling are implemented.
   b) If no detectable analytes are found in samples from an area with a low likelihood of contamination, then re-sampling is not undertaken.
   c) Reference the QAPP or similar quality-control document, to govern the sampling process including data verification and validation.

6) Establish statistical decision-making objectives, action levels, DCGLs, MDCs, and grid size. This step also includes the establishment of critical statistical parameters, that which define action levels for decision-making:

   a) If radiological analyses of samples show no detectable levels of analytes of concern, then the frequency and scope of sampling and analysis may be reviewed and curtailed.

   b) Statistical limits for decisions must be specified to control errors within defined, tolerable limits. A standard approach of 0.05 for both an "• error" and "• error" are commonly used (refer to Section 5.3.2).

7) Establish a continuous review and evaluation process to critique techniques and results in an on-going basis, and implement corrective actions, as required, to optimize designs for obtaining data.

Residual radioactivity is generally contained within systems and components. Because these are usually enclosed and plants typically assess system radioactivity on a regular basis, system characterization is more straightforward than that for small amounts of radioactivity that may inevitably escape from systems to the environment (as a result of normal plant operational tasks). Their concentrations may not be low relative to DCGLs. Extensive characterization work to examine and quantify the extent of the contamination may be required.

One important concept that must be considered during DQO development is the likelihood of environmental transport processes that may affect contaminants. For example, subsurface leaks that enter ground water may diffuse and transport contamination into the subsurface. Licensed releases to surface water bodies may cause radioactivity to accumulate in sediment. Licensed atmospheric releases may have deposited radioactivity on site land areas.

Although some components may be affected more than others, this knowledge is key to a thorough site evaluation. Physical and chemical processes can also be a factor in transport. When DQOs are fully integrated into the processes of planning, a thorough picture of site status and project scope can be assembled.

The DQO process provides flexibility at all stages of the survey process, including planning, implementation, and assessing compliance with the release criteria. As stated above in Step 7, the DQO process is one of iteration, involving the review and inclusion of new data and information into the development of the FSS. The NRC has identified difficulties in the development of DQOs by licensees who have not optimized the DQO process to their advantage. This includes dependence on initial characterization data, coupled with a resistance to incorporating new data, as it becomes available (Reference 2). Optimization of the DQO process may avoid significant time delays (see information added to MARSSIM logic in Figure 2-1 and Section 2.2.1.1).

2.2.3 Site Conditions: Geology and Geohydrology

Transport, as well as static conditions that affect the site environment will need to be based on detailed geologic and geo-hydrologic information about the site. The site FSAR and its supporting documents will contain basic information to initiate this part of the project. However, close attention to project work by individuals qualified in these fields will support the conclusions for media of concern in natural systems such as:

- Soils
- Bedrock
- Ground water
- Sediment

Understanding the conditions of these natural systems can be useful in the preparation of suitable field procedures and conduction of proper sampling and analytical work. Consultation with individuals possessing detailed knowledge of site environmental conditions will ensure a more efficient investigation and enhance results.

2.2.4 Preparing Sampling Procedures and Plans

In general, sampling procedures required for site characterization need to address:

- Project organization structure
- Sampling/Scanning for each medium of concern
- Sampling plan preparation
- Characterization and remediation sampling
- Sample chain of custody documentation
- Quality Assurance/Quality Control
- Health and Safety
- Database compilation

Specific sampling procedures are needed to address:

- Field sampling (surface soil, subsurface soil, sediment, ground water, concrete, building materials, etc.)
- *In situ* monitoring
- Database construction, management, and maintenance (including input timeliness, verification and validation)
- Chain of custody (sample packaging, labeling, documentation of transport)
- Data analysis and reporting

The general administrative procedure (Section 2.1.3) should list all procedures for site characterization as well as other project procedures that impact characterization work. It should explain the interactions required for performing the work, including organizational interfaces. It is very important to note that considerable craft resources will be required during certain periods of characterization. Such crafts include carpenters, pipe fitters, and laborers. These activities should not be underestimated during budget projections. Craft supervision with site-specific experience should be involved at the earliest planning stages to ensure efficient and accurate scheduling.

Sampling plans are work packages written for a specific location. They should address and document a number of practical elements:

- The purpose and scope of any sampling work
- The relevant historical data references
- Potential presence of known contaminants
- Planned locations of samples
- Method for identifying sample location (survey measurements)
- Depth of samples or sample segments
- Basis for sample locations and depths
- Methods for sampling and QC samples
- Methods for any special handling or treatment of samples
- Area-specific contamination control methods
- Identification of analytes
- Identification of an appropriate analytical laboratory
- Calibration and control checks for any field instrumentation used
- Field disposition of samples
- Sample chain of custody requirements
- Sample descriptions
- Drawings or maps locating embedded pipe or components
- Area preparation requirements
- Grid for determining locations of samples
- Health & Safety and job hazard information
- Radiological hazards and requirements
- Security access issues
- Facility operations issues that might impact sampling work or its results

Sampling procedures are designed specifically for the media and analytes of interest. Many standard references exist for such procedures (References 18 and 19). These procedures should not preclude screening samples, which are samples taken without full or standard procedures for informational purposes. However, analysis of such samples needs to be designated as such in the characterization database and not used for formal or reporting purposes.

Remediation procedures that address sampling will provide documentation to distinguish between a sample that represents material removed and material left in place. MARSSIM, Section 5.4, discusses this under the topic of Remedial Action Support Surveys.

Documentation from the electronic database should be used in the analysis of the data. The database does not have to be elaborate, but its accuracy and security is imperative. Input to the database (Section 2.3.5) should reflect the remediation sample type (e.g., whether material is removed or left in place).

An example sampling plan is provided in Appendix A.

2.2.5 Analytes of Concern

All possible analytes should be initially considered. The neglect of an analyte (either radiological or non-radiological) creates the risk of needing to re-sample or to re-analyze, a costly activity that can also make a project appear to be inadequate. Using a comprehensive list and the process of elimination, provides the best means to designate analytes of concern. Justification for elimination of any analyte is thus considered and documented.

Sources that will provide needed input to this process for radiological samples include existing analyses from:

- Plant waste shipments (e.g. evaporator bottoms),
- REMP samples,
- Plant process streams (e.g., coolant), and
- Systems

Some uses of materials that may produce non-radiological contaminants are common at large industrial facilities. A site characterization program that addresses this possibility in the above context is prudent. A starting point for such analysis would be the EPA's Title III List of Lists (Reference 20), a consolidated list of chemicals subject to RCRA and Section 112 (r) of the Clean Air Act. Elimination of specific substances or groups of substances can be made, simply by documenting that the material never existed on the site. Such investigation should be supported by the HSA as a primary reference.

A broad spectrum of initial ground water analyses will provide a means to confirm the absence of many potential analytes. At many sites, ground water can act as a screen to catch and disperse contaminants, such that, for scoping purposes, finding the specific location of a release is not required.

2.2.5.1 Provide for assessment of HTD radionuclides

The “hard to detect” radionuclides (HTDR), such as Fe-55 and Ni-63, are understood to be not readily detectable by field instruments. HTDRs are not necessarily hard to detect in the lab using special analytical laboratory techniques (often referred to as 10 CFR Part 61 analyses), thus, wet chemical analyses are required (such testing is usually destructive). Furthermore, the small volumes tested can create uncertainty about results due to non-homogeneous distributions within a sample. These analyses are also expensive and time consuming. Thus, both time and resources can be saved if predetermined (surrogate) relationships are established between the HTDRs and gamma emitting nuclides, such as Co-60. However, justification for the use of surrogate ratios is needed.

In recently submitted LTPs, the NRC staff review has identified sections where site characterization data are incomplete (Reference 2). One such example of deficient data is that pertaining to the absence/presence of transuranic (TRU) alpha-emitting nuclides and HDTRs (e.g., ^{3}H). TRU includes primarily longer-lived radionuclides, such as: ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, ^{241}Am, ^{237}Np, ^{242}Cm, ^{243}Cm, and ^{244}Cm (Reference 21). Although some LTP submittals report fuel clad failure history, they do not provide adequate data to support the absence/presence of these TRUs. Thus, sufficient detail is needed, such as prior Part 61 results and/or new laboratory analysis of samples, to confirm the absence/presence of TRUs and HTDRs (Reference 2).

2.2.5.1.1 Bulk Media

For the purpose of site characterization of bulk media (e.g., asphalt, concrete, sediment, soil), samples may be routinely analyzed using gamma spectroscopy, with only a few determinations made for HTDRs. This requires that a surrogate relationship be established between gamma-emitting nuclide concentrations and HTDRs. The basis for such an association is described in NUREG/CR-5849, Appendix A (Reference 22), where it prescribes the amount of radioactivity that constitutes a significant dose as follows:

> *"For sites with multiple radionuclides, only those radionuclides remaining at the time of license termination, which would contribute greater than 10% of the total radiation dose from all contaminants or which are present in concentrations which exceed 10% of their respective guideline values (GLVs), need be considered as significant contaminants."*

As a practical approach, the selection of samples with the highest concentration of gamma-emitting nuclides might be analyzed to the lowest practical MDC for the HTDRs. The assumption is made that the samples with the highest concentrations of gamma emitting nuclides would also contain the highest concentration of HTDRs.

2.2.5.1.2 Surface Contamination

Surface contamination DCGLs for detectable beta-gamma emitting nuclide concentrations may be developed for specific areas based on previously determined radionuclide distributions. Surface contamination guideline values are based on detectable, decay corrected beta-gamma emitting nuclide concentrations that incorporate the HTDRs. Title 10 CFR Part 61 analyses may be performed on composite smear samples representing major areas of a contaminated surface. The locations sampled should be taken from areas with a high potential for residual radionuclide contamination. Consideration should be given for providing a number of 10 CFR 61 sample batches to attempt to establish radionuclide mixes for identifiable waste streams and media.

Radionuclide composite distributions may be used to provide an effective DCGL, which accounts for the fraction of the distribution that will not be detectable with a radiation survey instrument (i.e. HTDR). One method of accounting for the HTDRs is to determine an effective DCGL value for the survey unit based on the detectable fraction. The radionuclide activities may first be decayed to the desired survey date with the decayed fraction determined as follows:

$$f_n = \frac{DA_n}{DA_{total}}$$

Where,

f_n	=	Decayed fraction for each radionuclide
DA_n	=	Decayed activity for each radionuclide
DA_{total}	=	Total decayed activity in sample

The effective DCGL is then calculated by the following:

$$DCGL_{eff} = \frac{F}{\frac{f_1}{DCGL_1} + \frac{f_2}{DCGL_2} + \ldots \frac{f_n}{DCGL_n}}$$

Where,

$DCGL_{eff}$	=	Effective DCGL value which accounts for the HDTRs
F or	=	Total decayed fraction for detectable radionuclides; Σf_n for detectable radionuclides only
$DCGL_n$	=	Radionuclide specific DCGL

2.2.6 Media of Concern

Most of the media of concern are readily evident from historical site data. Further determination of specific media might be included as a standard approach in scoping work. Media that are likely to be included are:

- Surface soil and paving materials
- Subsurface soil
- Ground water

- Concrete
- Building materials and surfaces
- Systems and components

Other uncommon or less recognizable media that have been identified as needing characterization have included:

- Paint
- Roofing materials
- Septic leach field materials
- Soil and construction debris from plant construction
- Sediment in water bodies used for release of licensed liquid effluents
- Vegetation (in particular, root mass of dense growth)
- Air or liquid filter media
- Air or liquid collection points (e.g., weir walls where low levels of contamination in water can accumulate and concentrate over time)

The HSA should be carried out with a broad concern for this aspect of characterization.

After system piping and components, the most significant media to be characterized at commercial nuclear power reactor sites include soil, ground water, concrete, building materials, and sediment.

Unique media requiring characterization may exist at any site. For example, radiologically contaminated septage can place suspicion on the soil that comprises the septic leaching field. Soil fill generated from new construction at nuclear power plant sites during operation may represent material on site that was governed by release criteria different from current standards. Subsurface leaks can create contaminated soil at a depth that is not detectable using surface techniques. Licensed releases from cooling water discharges or gaseous releases from a plant stack may accumulate due to natural actions that can concentrate such radioactivity.

An essential part of characterization studies is the detailed definition of the specific properties and nature of each medium. Such properties may control the presence and transport of radionuclides in environmental media. Sections 2.3.4.1, 2.3.4.3.1, and 2.3.4.3.8 illustrate examples of relevant conditions for soil and concrete.

2.3 Data Collection

2.3.1 Scoping and Background Surveys

The HSA contributes directly to the assessment of the scope of work required for a site. The second and more detailed step is the performance of a scoping survey (Reference 1, Section 4).

It is a preliminary round of investigation that furnishes a starting point for the comprehensive characterization investigation. The resultant data may be eventually included as characterization information.

As shown in Figure 2-1, scoping results should provide data that allow for classification of MARSSIM Class 3 areas, those areas unlikely to contain analytes of concern. This phase should also allow for the delineation of boundaries that identify any non-impacted areas of the site or any potential off-site migration of contamination.

Results of the scoping survey are typically in the form of characterization data used to prepare the project DP. In that document, these results should be described as preliminary, subject to confirmation, and not conclusive. A comprehensive summary, however, will provide a useful milestone report for a characterization project.

Background surveys performed early in the characterization program have the disadvantage of lacking extensive site-specific sampling experience with the media of concern. If possible, this work should be planned for times when sampling procedures and protocols are in place so that the sampling methods used are the same as those used in characterization.

References 22 and 23 provide thorough discussions of the issues related to this component of the work. For typical commercial nuclear power reactor sites, significant radionuclides for which background investigations may need to be considered are Cs-137, Sr-90, and H-3. Background may be ignored in some cases as being insignificant or simply to provide a conservative approach.

Reference 22 and Sections 4.5 and 5.5 of Reference 1 discuss methods to select background reference areas and determine the number of samples to be collected, respectively. Also, the EPRI report, “Considerations for Determining Background Radiation Levels in Support of Decommissioning Nuclear Power Facilities” (Reference 24), is a good technical reference document addressing background considerations.

2.3.2 Establish DCGLs

The license termination project will have a single primary objective: to fulfill all the necessary requirements for termination of the plant’s license. Related requirements are driven by the overall dose criteria of 25 mrem/yr (0.25 mSv/yr) (10 CFR 20 Subpart E). State regulations may also exert influence on this aspect of the project.

Once this project commitment is made, the next effort is to establish DCGLs. DCGLS are derived levels generally based on a site model using a dose modeling code (see Chapter 3) and refer to average levels of residual radioactive contamination above background. These concentration limits for each analyte and media (e.g., soils and buildings) of concern represent the release levels for building surface areas (e.g., dpm/100 cm^2) and material surface soil volumes (e.g., pCi/g). These DCGLs can be determined for both uniformly distributed and isolated areas of elevated residual radioactive contamination through the use of regulatory guidance (based on default values) or site-specific dose modeling analyses. Characterization data are a primary source of input to these codes (refer to Section 2.5.2) providing site-specific soil, ground water and other media radionuclide concentrations.

A derived concentration guideline level can be categorized as either a $DCGL_W$ or a $DCGL_{EMC}$. The $DCGL_W$ is a concentration level based on the assumption of uniformly distributed residual radioactivity over a large area, such as a survey unit (Reference 4, Section 2.5.2.1). The subscript "W" refers to the Wilcoxon Rank Sum test. This statistical test is recommended by MARSSIM for determining if residual activity exceeds the DCGL, where the radionuclide of interest is also present in background. MARSSIM also uses the DCGL subscript "W" for the Sign Test, which is a statistical test employed when the radionuclide of interest is not present in the background (Reference 1, Section 2.2, & 2.1.5.2). Again, the objective of this statistical test is to demonstrate that the median concentration of residual radioactivity in the survey unit is less than the $DCGL_W$.

When performing survey measurements, areas within a survey unit may be found to exhibit elevated concentrations of residual radioactivity, which exceed the $DCGL_W$ for the survey unit. Though they exceed the limit for the survey unit, they may still comply with the DCGL for a smaller isolated area of contamination. To demonstrate this compliance, a correction factor, or area factor, is employed to develop a new DCGL that sets a higher allowable limit for the smaller area (Reference 1, Section 2.5.3). This new DCGL, called a $DCGL_{EMC}$ (EMC refers to Elevated Measurement Comparison), is calculated by the following equation:

$$DCGL_{EMC} = \text{Area Factor} * DCGL_W$$

The area factor shown above is a multiple by which the concentration of residual radioactivity within the elevated area can exceed the $DCGL_W$ and still demonstrate compliance with the release criteria. The determination of area factors can be accomplished through the use of an exposure pathway modeling computer code. For example, RESRAD provides the necessary pathway modeling approach to determine area factors for a given area of elevated concentration. The user can input a unit concentration (1 pCi/g) for a given radionuclide of concern and, by using RESRAD in default mode, can determine the total dose received by an individual for the default contaminated area size of 10,000 m^2 for surface soil and the default of 36 m^2 for structures. Once this value has been obtained, the user can then run RESRAD for smaller contaminated area values, to calculate the corresponding dose received in each case. Changing both the area of the contaminated zone and the length of the contaminated zone parallel to the aquifer flow, while leaving all other input parameters (default values) constant, allows the user to determine how the reduced area impacts the dose. Thus, the smaller area doses can then be compared to the default area (10,000 m^2) dose. This comparison allows the user to determine the area factor for the smaller contaminated area. The equation would be set up as follows:

$$\text{Area Factor} = \frac{T_D}{T_A}$$

Where,

T_D = the total dose determined for the default contaminated area (10,000 m^2) with a radionuclide concentration of 1 pCi/g (all other parameters set to default).

T_A = the total dose determined for a smaller contaminated area with a radionuclide concentration of 1 pCi/g (all other parameters set to default).

In using the computer dose modeling code to calculate the total dose for a smaller contaminated area, the result will be lower than the total dose for the larger area. This lower dose is a result of using an equivalent residual radioactivity concentration paired with a smaller size for the contaminated zone (smaller source). It should be noted that the above discussion of the $DCGL_{EMC}$ is only considered for survey units designated as Class 1 (survey units classified as Classes 2 or 3 should not contain any areas of elevated radioactivity).

In Tables 2-1 and 2-2, example area factors for several areas of elevated concentration have been provided for both outdoor and indoor survey units, respectively (Reference 1, Section 5.5.2.4). These tables were calculated using pathway exposure modeling provided in RESRAD 5.61. For example, Table 2.1 shows that the Area Factor for a 10 m^2 land area contaminated with Cs-137 will be 2.4. This means that the dose from a 10 m^2 area contaminated with CS-137 will be a factor of 2.4 lower than a 10,000 m^2 area contaminated at the same concentration level. The $DCGL_{EMC}$ for this small area would, therefore, be a factor of 2.4 higher than the $DCGL_w$, or $DCGL_{EMC} = 2.4 \times DCGL_w$.

Table 2-1
Examples of Outdoor Area Factors[1]

Nuclides	Size of Elevated Concentration Area								
	1 m^2	3 m^2	10 m^2	30 m^2	100 m^2	300 m^2	1000 m^2	3000 m^2	10000 m^2
Am-241	208.7	139.7	96.3	44.2	13.4	4.4	1.3	1.0	1.0
Co-60	9.8	4.4	2.1	1.5	1.2	1.1	1.1	1.0	1.0
Cs-137	11.0	5.0	2.4	1.7	1.4	1.3	1.1	1.1	1.0
Ni-63	1175.2	463.7	154.8	54.2	16.6	5.6	1.7	1.5	1.0
Ra-226	54.8	21.3	7.8	3.2	1.1	1.1	1.0	1.0	1.0
Th-232	12.5	6.2	3.2	2.3	1.8	1.5	1.1	1.0	1.0
U-238	30.6	18.3	11.1	8.4	6.7	4.4	1.3	1.0	1.0

[1] MARSSIM Table 5.6

Table 2-2
Examples of Indoor Area Factors[1]

Nuclides	Size of Elevated Concentration Area					
	$1\ m^2$	$4\ m^2$	$9\ m^2$	$16\ m^2$	$25\ m^2$	$36\ m^2$
Am-241	36.0	9.0	4.0	2.2	1.4	1.0
Co-60	9.2	3.1	1.9	1.4	1.2	1.0
Cs-137	9.4	3.2	1.9	1.4	1.2	1.0
Ni-63	36.0	9.0	4.0	2.3	1.4	1.0
Ra-226	18.1	5.5	2.9	1.9	1.3	1.0
Th-232	36.0	9.0	4.0	2.2	1.4	1.0
U-238	35.7	9.0	4.0	2.2	1.4	1.0

[1] MARSSIM Table 5.7

There are instances where the "W" and "EMC" DCGLs should be considered simultaneously. Such a situation can occur when the presence of both uniformly distributed and elevated areas of contamination are found to exist within the same survey unit. Under those conditions, the unity rule (Reference 1, Section 4.3.3) can be used to demonstrate compliance with the release criteria. The unity rule equation (Reference 1, Equation 8-2) is shown here for the above conditions:

$$\frac{C_A}{DCGL_W} + \frac{\text{Average concentration in the elevated area} - C_A}{DCGL_{EMC}}$$

Where,

C_A = average residual radioactivity concentration in the survey unit

If the above result is less than 1, then the survey unit meets the release criterion (Reference 1, Section 8.5.2). If, however, there are multiple elevated areas, then an additional term for each elevated area should be added to the above equation for determination of compliance.

DCGLs should be estimated as early as possible in the characterization process in order to determine an efficient sampling plan based on a realistic radiological target. The DCGLs will also be used in the License Termination Plan.

2.3.3 Laboratory Options

The establishment of dose criteria and DCGLs will allow for the determination of the necessary requirements for laboratory analysis of characterization samples. Criteria for choosing an analytical laboratory include a number of important laboratory attributes. A list of information useful for laboratory screening is as follows:

- Organizational Chart with the number of staff in each group and the resumes of key personnel (lab director, QA manager, technical group managers).
- Facilities (date lab was built, square footage, major equipment list, type of Lab Info Mgmt System (LIMS).
- Certifications.
- Copies of last two EPA Grading results (Water Supply (WS) and Water Pollution (WP) results including any responses to the state certification authority addressing those results).
- Copies of any Performance Evaluation (PE) results required by programs for certification maintenance.
- Copy of standard laboratory reports for analytes of concern in media to be tested.
- Copy of the laboratory's QA/QC plan.
- List of services offered with standard fee schedule.
- References, especially any relevant to the project or its investigation character.

Sections 7.3 and 7.4 in MARSSIM (Reference 1) also contain a broad range of useful information with respect to interfacing with laboratories.

2.3.4 Sampling and Laboratory Analysis

Several matters need to be considered prior to the initiation of characterization sampling. This includes the sampling schedule, re-sampling criteria, sample location grid establishment, final survey sampling procedures, and issues related to individual media.

The goal of license termination will require a comprehensive planning and scheduling effort, including such tasks as systems removal, building demolition, and regulatory submittals. Site characterization will provide unique input into these plans. A decommissioning project causes a site to be busy with many tasks beyond site characterization, as they will sometimes control the availability of locations for characterization. This limits the availability of areas for characterization work and can also inadvertently reintroduce radioactivity into a characterized area. Coordination of work through an overall project schedule is vital to both control and organization of site characterization.

The site characterization schedule may require logistical support, such as access to radiological areas with attendant radiological protection support or to areas unavailable due to more encompassing tasks. In addition, the ability of laboratories to process samples may have an effect on sampling schedules. Sampling done by areas, defined by scoping as Class 1, 2 or 3, will also help to systematize a schedule. The development of a permanent site grid, to establish sample locations, is highly recommended.

Individual media may have sampling requirements that demand specific consideration. For example, ground water wells should be sampled periodically over the course of the project, as this medium represents a dynamic system, to provide assurance that the decommissioning work itself is not introducing additional contamination into the environment. Seasonal variations of components in ground water should also be assessed, as this is a standard approach for evaluating drinking water and ground water systems at contaminated sites.

Determining the boundaries of the areas and survey units may require particular attention, especially where boundaries separate:

- Areas assumed to not be impacted by plant operation, and
- Class 2 and Class 3 areas (Reference 1).

Site characterization investigations may identify some locations where more sampling is needed to fully characterize a medium (Figure 2-1). Whether the result of scoping or characterization, conclusions regarding some analytical results will call for more sampling to be done. Typical objectives of re-sampling could be:

- Resolve unexpected or unsatisfactory results (high or low)
- Confirm that unexpected analytes are present
- Ascertain limits (vertical or horizontal) of an area not defined by initial sampling
- Confirm results for important locations

In some cases site history might be re-examined for overlooked evidence that might provide a basis for findings (refer to Section 2.4.7).

Sampling data gathered for characterization purposes can be used as part of the final survey data. This can provide an advantage to the project, however, it requires sufficient planning to ensure that all requirements for the final survey samples are fulfilled.

A reference coordinate system for locating samples needs to be established to document sample locations. All sample collection data should reference this location system. Given that plant structure drawings play a key role in any sampling at a site, use of the plant construction grid may provide the most practical means for fulfilling this requirement.

2.3.4.1 Surface soil and asphalt

The most readily sampled natural medium on a site is surface soil. For DCGL determination, Reference 1 defines surface soil as extending to a depth of 6 inches (15 cm) below the surface. As a practical matter, "surface soil" might be considered as extending to depths of 12 inches (30 cm) (Reference 8) (not for DCGL determination). However, a different basis for dealing with investigation of soil needs to be considered when dealing with soil below this level (see subsurface soil, Section 2.4.4.3).

Soils on nuclear power plant sites tend to include a large amount of sandy, sometimes gravelly fills, used for construction purposes. This can provide a relatively homogeneous condition for sampling programs and scans done within site areas. It should not be assumed that soil areas are homogeneous. Variations due to deposition, hydrogeology, etc. are common, making strategic sampling imperative for accurate characterization. In addition, the range of natural soils that may need to be investigated may vary widely. Thus, a program that provides soil sample description as part of sample documentation can greatly enhance the results. The plant Final Safety Analysis Report (FSAR) can provide important guidance for this effort.

Natural variations in soil include: mineralogical/organic components, grain size, grain size distribution, and texture. These variations may result from natural processes that can segregate or alter soil composition and character (this variation can affect the soil affinity to bind with radionuclides). A comprehensive handbook for this information, the Unified Soil Classification System (USCS) (Reference 19), is a preferred method for soil sample description.

Frequently, surface soil is covered with asphalt or vegetation. Asphalt has a finite porosity and some contaminants will pass through it. Radiological substances will very often accumulate just below asphalt at the top of the soil layer, due to the inherent properties of both substances. Care should be exercised when identifying and evaluating this condition. Vegetative cover, or more correctly its root mass, may also provide a significant trap for migrating radionuclides. This horizon may require a separate evaluation.

2.3.4.2 In-situ methods/gamma scans

In situ gamma measurements can be performed for discrete locations, including structural surfaces and ground areas. A comprehensive scope of ground surface data for all radiological areas should be within the scope of characterization studies for most sites. Recent advances in techniques have made this tool more economical, although contaminant geometry issues require careful consideration in use of this tool as a quantification device for site final release. Reference 18, Volume 1, Section 3, provides a comprehensive discussion in the use of *in situ* methods. Some specific project results are provided in Reference 25.

2.3.4.3 Subsurface and sub-foundation soil

MARSSIM does not currently address subsurface radioactivity. Thus, each licensee should attempt to evaluate, remove, and provide a final assessment of any subsurface radioactivity on an individual basis. Obviously, the existence of sub-foundation contamination can have enormous cost impact for buildings that potentially could be left intact. Therefore, extensive evaluation of sub-foundation contaminants is warranted.

Although subsurface radioactivity levels above site DCGLs might be considered for removal, arguments have been made that excavation of such material would also include the materials surrounding the elevated area, thus significantly diluting the overall radioactivity concentration (References 6, 27 and 28).

Reference 8, Appendix E, also describes how allowances for buried radioactivity is accounted for:

> *"The $DCGL_W$ may be based on the assumption that the residual radioactivity may be excavated some day and that mixing of the residual radioactivity will occur during excavation. When the subsurface residual radioactivity is mixed and brought to the surface, most of the dose pathways will depend only on the average concentration."*

This reference further recommends that the extent of subsurface radioactivity, including depths, be determined with:

> *"the final site survey…performed…taking core samples to the measured depth of the residual radioactivity. The number of cores to be taken is the number N required for the WRS or Sign test, as appropriate. However, the mixing volume assumed in the scenario may require a larger number of core samples. There is no adjustment to the grid spacing for the elevated measurement comparison because scanning is not applicable. The core samples should be homogenized over each 1 meter of depth. Then the appropriate test (WRS or Sign) is applied to the set of samples. In addition, each individual core sample is also tested against a site-specific volumetric elevated measurement comparison.*

Overriding considerations would include the ease with which removal can be achieved, soil conditions, and the relative position of the residual radioactivity, with respect to the ground water table. In any case, site characterization will always have to assess existing conditions to provide a data point for initial assessment.

Reference 8, Appendix E, provides guidance indicating that subsurface investigation may be needed only when suggested by the HSA results. Experience suggests that subsurface radioactivity should also be investigated wherever significant concentrations of surface activity are found.

2.3.4.3.1 Soil boring methods for subsurface soil characterization

Soil borings are a standard approach for obtaining subsurface soil samples. Due to their expense and limited lateral reach, they might be considered under specific conditions, where:

- Higher concentrations of radioactivity have been identified or inferred
- Significant depth to known radioactivity or other physical configuration limits excavation
- Ground water monitoring will be required and a boring to install a monitoring well will be required in any case
- Locations where depths below the ground water table require sampling

A modified method of split-spoon sampling (References 28 and 29) has provided good results, when precautions are taken to avoid cross-contamination. This can be inherent to some soil boring approaches, such as with remote sampling methods.

Split spoons are metal tube soil samplers split lengthwise along a grooved joint with a hardened cutting shoe at the bottom. The sampler is driven into soil by a weight dropped 30 inches (76.2 cm). Hammer blows are tabulated in a way that provides a measure of soil strength. Split-spoon sampling can provide a series of samples to help assemble a comprehensive soil profile for a given location. Standard split spoons are 2 inches (5.08 cm) in diameter, 18 to 24 inches (45.7 or 60.9 cm) long, and can be readily dismantled to remove soil samples. However, 3-inch (7.62 cm) diameter spoons, which are 24 inches in length (60.9 cm), should be used to obtain samples of suitable volume for radiological analyses. Care is required to ensure that cross-contamination potential is minimized.

2.3.4.3.2 Test pits

An inexpensive, practical, and productive means of shallow subsurface exploration for soil contaminants is by the use of test pits, dug with a backhoe. Standard procedures (Reference 30), coupled with appropriate precautions to prevent cross-contamination, are sufficient to provide reliable data. Several locations can be sampled quickly with sampling depths controlled by:

- The “reach” of the machinery used,
- Availability of equipment that ensures safe entry into an open trench, and/or
- Depth to the ground water table, below which caving of pits may be expected.

Test pits will create excavated soil and cause removed-soil to become intermingled. As a result, any soil removed should be controlled until a determination is made as to the concentration(s) of the analytes of concern. Separate sampling of the excavated soil may also be considered.

2.3.4.3.3 Remote sensing methods

Landfills or subsurface disposal areas associated with facilities may demand that the absence of buried radiological materials be confirmed. Remote sensing methods can be used to identify subsurface materials. By locating such items as underground tanks, buried drums, or drain pipes, the search for potential residual radioactive contamination can be effectively narrowed. Methods such as ground magnetic and electromagnetic conductivity (terrain conductivity) surveys, and ground penetrating radar can identify buried metallic objects that may potentially contain radioactive contamination. Reference 31 provides a comprehensive description of such methods.

2.3.4.3.4 Sub-Foundation/Sub-Floor Soil Samples

Experience has shown that for radiological areas within nuclear power plant structures, sampling soil beneath concrete floors (boring through the foundation to reach the soil beneath) where standing water occurred (intentionally or unintentionally) can produce samples with concentrations of radioactivity as a result of migration through the concrete floors. Soil beneath unlined concrete sumps, pipe chases, pits, or cavities are thus suspect. Methods similar to those used for outdoor subsurface soil and ground water sampling can be employed.

2.3.4.3.5 Sediment

Sediment is merely soil that is beneath and/or saturated with water. It may tend to be more uniform in texture than soils due to actions caused by water transport. However more significant transport mechanisms for both sediment and associated water have potential for moving residual radioactivity. Transport mechanisms may either disperse or concentrate radioactivity, depending on the specific radionuclides and sediment character. Thus, a separate evaluation of any sediment may be warranted for characterization. Sediment transport and deposition rates should be considered in any such study.

In cases where licensed liquid effluent releases to water bodies that contain sediment have occurred, the possibility that contaminated material may be detected by characterization studies must be considered.

2.3.4.3.6 Surface Water

The effects of nuclear power plant operation can be sought in any nearby water body. Thus, water from surface water bodies that is used for plant cooling purposes is normally the object of REMP sampling. Thus, REMP surface water sampling locations are most likely to contain residual radioactive contamination.

Since the solubility of most radionuclides is very low, the likelihood of finding radioactivity in the water of surface water bodies is commensurately low.

In the NRC staff review of recently submitted LTPs, the details of site characterization of surface water should be such that the licensee has sufficiently described the type and extent of the radioactive contamination present (Reference 8, Section 4.5). The following citation from NUREG-1727, Section 4.5, outlines some of the information requirements for descriptions of surface water contamination:

- *A list or description and map of all surface water bodies at the facility that contain residual radioactive material in excess of site background levels;*
- *A summary of the background levels used during scoping or characterization surveys; and*
- *A summary of the radionuclides present in each surface water body and the maximum and average radionuclide activities in picoCuries per liter (pCi/l).*

2.3.4.3.7 Ground Water

Ground water represents a mass of water underlying a site. It can capture or dissolve many introduced contaminants. Its flow will potentially carry contaminants across or out of a site at variable rates.

Ground water data obtained through operational environmental monitoring, such as REMP sampling done for nuclear power plants, may not be sufficient for site characterization purposes or for the provisions of supporting dose assessments. The data collected from REMP programs can provide valuable insight for site characterization. However, the data tend to be insufficient to provide the necessary details that the NRC staff requires in developing an understanding of the site groundwater conditions (Reference 2). An example of the scope required for such data is the description of movement and extent of radioactive contamination in the ground water (Reference 8, Section 4.6).

Ground water sampling is done by means of designated monitoring wells (Reference 31). State environmental agencies may have monitoring well design regulations that should be consulted to determine potential applicability. The location of monitoring wells should be chosen to take advantage of ground water flow direction and rates. This requires an understanding of site geology and hydrogeology.

Given properly designed, constructed, and maintained observation wells, sampling for most radiological analytes is not highly complicated. However, sampling for some non-radiological components, such as volatile organic components, can require detailed methods. A standard approach to determining the concentration of regulated contaminants in drinking water is by taking quarterly samples over a period of at least one year. The planned scope of sampling for any monitoring well should accommodate this procedure.

Ground water monitoring will also provide data to assess the impact of decommissioning work itself. The characterization program should address this too, with sampling scheduled and located to assess any consequences due to tasks that decommissioning entails.

The most likely nuclear power plant radionuclide released to ground water is tritium (^{3}H). While this radionuclide is often present, it is typically in concentrations well below drinking water standards. Its presence can also provide a tracer that will clearly identify the flow path of ground water for the site and thus for other contaminants.

2.3.4.3.8 Concrete

Concrete has the capacity to accumulate radioactivity due mainly to its contact with water containing radioactivity. For sumps, pits, and pipe chases (common in plant buildings that are intentionally or unintentionally subject to filling with water containing radioactivity), this condition is expected. In addition to concrete's small but finite permeability, discontinuities, such as shrinkage, cracks, and construction joints, increase this tendency and may concentrate radioactivity.

Standard commercial coring techniques generally provide adequate results to evaluate concrete that is part of standard building structures. Investigations using cores sliced thin to provide depth profiles from a surface have been used to assess the thickness of contaminated concrete.

2.3.4.3.9 Other Building Materials, Surfaces (Interior and Exterior)

Radioactivity in building materials of nuclear power plant structures can be elusive. Even where sources are obvious, radioactivity can end up trapped in materials or found in unusual locations. For example, the concrete comprising a reactor cavity at a nuclear power reactor site was known to have had original construction joints where wood debris, left behind at the time of construction, provided a trap for radioactivity deep inside the cavity walls. This made concrete removal difficult and expensive. In addition, roofing materials can trap residual particulate radioactivity from gaseous releases made from a plant stack.

2.3.5 Database Compilation of Analytical Results

Analytical data may often be tabulated and interpreted in the field, on a preliminary basis soon after its collection. However, a means of formal analysis and reporting needs to be available to ensure that data are properly interpreted by qualified personnel with conclusions properly documented.

All analytical results should be compiled in an electronic database that allows permanent storage and easy, but controlled, access for data review and output. Data should be entered into the database on a timely basis. Data verification and validation, as described in References 1 and 32, Appendix N, will help to provide legally defensible data to support the project.

Input requirements for a given analysis might include:

- Lab Number
- Field Number
- Descriptive location at the site
- Specific location at the site
- Date collected
- Analytical laboratory name
- Sample dry or wet weight
- Soil description
- Soil type (USCS, Reference 33)
- Sample depth, elevation, or point of reference
- Sample type (information, characterization, remediation, etc.)
- Comments
- Site Coordinates
- Latitude, Longitude
- Sampler initials
- FSS survey area
- Radionuclide concentration
- Radionuclide reported laboratory error
- Radionuclide laboratory MDC
- Radionuclide flag for detectable concentration

- V&V flags (Verification, Validation, invalid data, contamination suspected, etc.)

2.4 Data Analysis

2.4.1 Area Classification Assessment

Area classification for the purpose of a MARSSIM final survey is based on the HSA and characterization data (Reference 1, Section 4.4). While classification of an area is based on a very conservative approach, a high degree of certainty regarding the potential for contamination provides significant project efficiency. While failed results from a specific final survey area can greatly impact a program, over-conservatism relative to the classification of areas can also be detrimental.

The confidence with which characterization data can be presented and interpreted is critical in ensuring that the initial step of the final survey process is properly implemented.

2.4.2 Data Compilation and Assessment

Site characterization data should provide a reasonable base in the compilation of data for performing dose assessments and, later, to provide input to the FSS. For example, site characterization data will provide a means for estimating the volume of materials that will be shipped offsite as radiological waste and estimating the potential for activity in plant systems or components left in place.

In the first example, the projection of waste volumes is a relatively straightforward task, provided a sufficient number of samples are taken. Note that characterization sampling should take into account optimization of waste packaging (e.g., segmentation plans for optimum package weights and dimensions, as waste costs can be 1/3 of the total D&D project budget). In the second example, site characterization will provide information used in the decision as to whether minimally contaminated plant systems components, or as a whole, should be left in place after license termination.

2.4.3 Inputs for Dose Assessments

Most site dose assessments will be performed using either RESRAD (Reference 34) and/or DandD Screen (see Chapter 3). Site-specific inputs to both these codes may be generated from site characterization data. The input data, provided either directly or indirectly by site characterization studies or results, can include:

- Geometry of contaminated zones
- Watershed extent
- Aquifer flow direction
- Elapsed time since radioactivity placement or release

- Concentrations of radionuclides
- Hydrologic and physical parameters for site soils
- Description of soil horizons relative to the likely range of distribution coefficients for radionuclides

Distribution coefficients are among the most complex data input to these codes. Site-specific measurements may be performed using the methods described in Reference 35.

In the course of providing inputs for the dose assessment, techniques and methods for site characterization should be assessed. Methods used for site characterization should be open to improvement and innovation, as they can often prove beneficial to the process. Advancements in sampling and analytical techniques are realized frequently. The final step of the DQO process (Section 2.2.2), which seeks to critique techniques and results in an on-going basis and implement corrective actions as required to optimize designs for obtaining data, provides a direct means to implement such new ideas and techniques.

When optimizing site characterization data, an assessment should also be performed to establish consistency with the HSA and determine if re-investigation is necessary. As discussed in Section 2.2.1, the HSA will be tested when characterization data are analyzed. Areas of contamination should show results consistent with their history. Where this is not the case, a reassessment of history is in order. Unusual results will sometimes prompt recollection of pertinent information that can reconcile such contradictory results.

2.4.4 Input to FSS work

Site characterization is required to provide essential information to be used as a basis for the final survey. Based on MARSSIM, a final survey will require the following elements, related directly to site characterization:

- Identified contaminants
- Established DCGLs
- Classified site areas by contamination potential
- Data to define each survey area and/or unit
- A reference coordinate system
- A documented null hypothesis (residual radioactivity exceeds release criteria)
- The mean concentration in the survey unit for setting a lower bound of the gray region (LBGR)
- Defined Type I (α) and Type II (β) decision areas
- The standard deviation (σ) for characterization measurements made in each survey unit
- The relative shift (Δ/σ) (generally between 1 and 3)
- The MDC for all analytical techniques

- The number of measurements for all survey units
- Maps for survey units are provided
- Specific survey documentation to be collected

These requirements define the interface between site characterization and final survey projects. Close coordination between these efforts can enhance project efficiency.

Site characterization data must provide residual radioactive contamination levels on the site for the final status survey. This includes data representing those levels that exist after the completion of all remediation work. However, the NRC recommends avoiding collection of additional data that is not pertinent to the design of the final status survey (Reference 2). Site characterization data, such as the characterization of those media or building surfaces that clearly exceed the applicable DCGLs, need not be considered in the LTP preparation. Furthermore, the process of obtaining such data may present both increased exposure to workers and higher costs. Post-remediation survey data for survey units may be used to obtain mean and standard deviation values for use in the Final Status Survey.

2.4.5 Remediation

Decommissioning work typically requires removal of systems and building materials and remediation of soils. Depending on site conditions, other media may also be remediated. Chapter 4 of this report provides a comprehensive summary of remediation issues. However, two aspects of soil remediation are addressed here due to their impact on site characterization work.

Analysis of soil samples that are taken to guide remediation will produce a final phase of characterization data for a given area. Expected results may allow for a less exhaustive initial characterization sampling scope for impacted locations. Analytical results for these remediation samples should be entered into a database for site characterization analytical results (refer to Section 2.3.5).

Remediation of soil to a given level of concentration of contaminants may still allow pockets of residual contamination beyond the extent of initial cleanup. A standard approach would be to remediate to a slightly lower standard than required to provide substantial assurance that the full extent of contamination has been removed.

2.5 Analysis and Presentation of Results

Clear concise reports describing site characterization analysis and results are needed to allow many project areas to assess their own needs and plans. These may take many forms; a generalized example of how these might be addressed is depicted in Appendix B. Reports must also be sufficient to address all regulatory requirements, as well as keeping management updated on project progress. Action item tracking, tied to periodic meetings, is a method that can provide a useful tool in a multi-faceted project such as nuclear power plant decommissioning. Accurate mapping and detailed graphics will increase the reader’s understanding.

2.6 LTP Input

2.6.1 Standard Format and Content - Reference 6

Provides a summary of the required input to the LTP that is based on site characterization work, as follows:

1) It prescribes measurements that identify maximum and average contamination levels and ambient exposure rate measurements of all relevant areas as follows:

   a) Identification of all locations inside and outside the facility where radiological spills, disposals, operational activities, or other radiological accidents/incidents occurred and that could have resulted in contamination of structures, equipment, lay-down areas, or soils (sub-floor and outside areas).

   b) A summary description of the original shutdown and current radiological and non-radiological status of the site.

   c) Site characterization with sufficient detail to allow the NRC to determine the extent and range of radiological contamination of:

      i) Structures

      ii) Systems (including sewer and waste management systems)

      iii) Floor drains

      iv) Ventilation ducts

      v) Piping and embedded piping

      vi) Rubble

      vii) Contamination on and beneath paved parking lots

      viii) Ground water and surface water

      ix) Components

      x) Residues

      xi) Environment

2) Identify the survey instruments and supporting quality assurance practices used in the site characterization program.

3) Provide sufficient detail for planning further decommissioning activities, such as:

   a) Decontamination techniques

   b) Project schedules

c) Costs

d) Waste disposition plan (optimization of segmentation, volumes, packaging, shipment, and disposal)

e) Dose assessments (including ground water)

f) Health and safety considerations

4) Provide description of remediation technique.

2.6.2 NUREG-1727 Standard Review Plan - Reference 8

Provides additional recommendations that may pertain to characterization input to the LTP for various site media, as follows:

Surface/Subsurface Soils

- Information relating to soil in the top 6 to 12 inches (15 to 30 cm) (surface) and soil below this level (subsurface).
- A list or description of all locations at the facility where surface/subsurface soil contains residual radioactive material in excess of site background levels
- A summary of the background levels used during scoping or characterization surveys
- A summary of the radionuclides present at each location, the maximum and average radionuclide activities, the chemical form of the radionuclide, and if multiple radionuclides are present, the radionuclide ratios.
- The depth of subsurface soil contamination at each location
- The maximum and average radiation levels…at each (surface soil) location
- A scale drawing or map of the site showing locations of residual radioactive material contamination in surface/sub-surface soil.
- A summary of the aquifer(s) and surface water bodies at the facility that contain residual radioactive material in excess of site background levels
- Summaries of the radionuclides present in each aquifer and surface water body and the maximum and average radionuclide activities.

Contaminated Structures

- A list or description of all structures at the facility where licensed activities occurred that contain residual radioactive material in excess of site background levels.
- A summary of the structures and locations at the facility that the licensee or responsible party has concluded have not been impacted by licensed operations and the rationale for the conclusion.
- A list or description of each room or work area within each of these structures.

- A summary of the background levels used during scoping or characterization surveys.
- A summary of the locations of contamination (i.e., walls, floors, wall/floor joints, structural steel surfaces, ceilings, etc.) in each room or work area.
- A summary of the radionuclides present at each location, the maximum and average radionulide activities, the chemical form of the radionuclide, and, if multiple radionuclides are present, the radionuclide ratios.
- The 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).
- The maximum and average radiation levels …in each room or work area.
- A scale drawing or map of the rooms or work areas showing the locations of radionuclide material contamination and radiation levels.

Contaminated Systems and Equipment

- A list or description and the location of all systems or equipment at the facility that contain residual radioactive material in excess of site background levels.
- A summary of the radionuclides present in each systems or on the equipment at each location, the maximum and average radionuclide activities, the chemical form of the radionuclide, and, if multiple radionuclides are present, the radionuclide ratios.
- The maximum and average radiation levels at the surface of each piece of equipment
- A summary of the background levels used during scoping or characterization surveys.
- A scale drawing or map of the rooms or work areas showing the locations of the contaminated systems or equipment.

Note that consideration should be given to whether or not systems or buildings will be removed prior to license termination before implementing the above requirements.

The NRC recommends that all site characterization work, intended to satisfy the necessary LTP criteria, be completed before its submittal. Sometimes LTPs are submitted in the midst of an on-going site characterization. These should possess enough flexibility to integrate potential changes. The LTP should demonstrate that a sufficient site characterization has been performed to assess the radiological conditions of all site media and to adequately support area classifications (Reference 2) (see Section 2.4.4).

3
DOSE MODELING

The U.S. Nuclear Regulatory Commission (NRC) has published a Subpart E to 10 CFR 20 that contains the radiological criteria for terminating a license. In addressing these criteria, dose modeling is performed based on the residual levels of contamination on the site (source term), the critical population group(s), and the applicable dose pathways. Computer codes have been developed by the NRC and DOE that implement these dose models based on defined critical population groups and pathways. The computer codes allow the radionuclide specific source term and selected site parameters to be input into the analysis. Results may be obtained either as allowable concentrations of radionuclides in soil or on building surfaces or in terms of risks or doses to the public.

This section of the report discusses the dose criteria applicable to license termination, the concept of critical population groups, and the computer codes that have been developed for dose modeling. The use of these codes as both a screening tool and in site specific analysis is covered. Guidance is also provided in the use of the codes as a tool for predicting the distribution of doses to the public (probabilistic dose modeling).

3.1 Site Release Criteria

The radiological criteria for unrestricted release of a site containing residual radioactivity are provided in 10 CFR 20.1402. The total effective dose equivalent (TEDE) to the average member of the critical population group (see Section 3.3) from residual radioactivity that is distinguishable from background radiation must not exceed 25 mrem per year (0.25 mSv/yr). This criterion includes doses that may result from residual radioactivity in groundwater and the drinking water pathway. The release criteria also require an ALARA assessment that demonstrates that residual radioactivity has been reduced to levels that are as low as reasonably achievable. The ALARA assessment is discussed further in Section 4.3 of this report.

The site release criteria apply to land areas and buildings. Buildings include equipment and systems that are attached to the building, such as lighting fixtures and laboratory benches. Equipment not attached to the building, such as office furniture, is not covered by these criteria.

Site release criteria are also provided for restricted release in 10 CFR 20.1403. These criteria require that the 25 mrem per year (0.25 mSv/yr) TEDE be met by providing for institutional controls that would limit exposure to the public. These criteria also require that the dose to the average member of the critical group be no greater than 100 mrem per year (1 mSv/yr) [or 500 mrem per year (5 mSv/yr) with additional requirements] if the institutional controls are no longer in effect. It should be noted that many site stakeholders (e.g., state regulator) are requiring more conservative TEDE limits. Projects should consider the possibility that more conservative limits may be implemented which will be more difficult to evaluate and attain.

Alternate criteria for license termination are also provided in 10 CFR 20.1404 for sites that will not meet the 25 mrem per year (0.25 mSv/yr) dose limit. These alternate criteria include a demonstration that doses to the public would be no greater than 100 mrem per year (1 mSv/yr) from all man-made sources of radiation other than medical.

3.2 Critical Population Group

The radiological criteria established within 10 CFR 20, Subpart E, sets limits on the annual total effective dose equivalent (TEDE) to the average member of the critical group. This critical group is defined in 10 CFR 20.1003 as "the group of individuals reasonably expected to receive the greatest exposure to residual radioactivity for any applicable set of circumstances". According to the ICRP, the critical group should be small enough to be relatively homogeneous with respect to those aspects of behavior that affect the doses received (Reference 36).

As required under 10 CFR 20.1302(b), expected doses are evaluated for this average member of the critical group. It is likely that doses to some members of the group could be higher than the average. However, due to the conservative assumptions used in the dose models, the actual doses received by members of the public will normally be lower than those calculated for the critical group. While it may be possible to actually identify the most exposed member of the public in some operational situations (through monitoring, time-studies, distance from the facility, etc.), identification of the specific individual that will receive the highest dose some time in the future (up to 1000 years) is impractical, if not impossible. Speculation on his or her habits, characteristics, age, or metabolism could be endless. The use of the "average member of the critical group" acknowledges that any hypothetical "individual" used in the performance assessment is based, in some manner, on statistical results from data sets (e.g., the breathing rate is based on the range of possible breathing rates) gathered from groups of individuals. While bounding assumptions could be used to select values for each of the parameters (i.e., the maximum amount of meat, milk, and vegetables ingested, possible exposure time, etc.), the result could be an extremely conservative calculation of an unrealistic scenario and may lead to unreasonable residual radioactivity levels.

By using the hypothetical critical group as the dose receptor, coupled with prudently conservative models, it is highly unlikely that any member of the public would actually receive doses in excess of those calculated. The description of a critical group's habits, actions, and characteristics should be based on credible assumptions. The information or data ranges used to support these assumptions should be limited in scope to reduce the possibility of adding members of less exposed groups to the critical group. Two critical population groups have been modeled in the computer codes that have been developed for dose assessment. The first is a group of resident farmers that occupy the land areas on the site, build homes, plant gardens, raise livestock, and drill wells for irrigation and drinking water (resident farmer scenario). The second is a group of workers that occupy the buildings left on site and use these building for occupational purposes (building occupancy scenario).

3.3 Computer Codes for Dose Modeling

Dose assessments are typically carried out through the use of site characterization, dose modeling, and other analytical tools to demonstrate compliance with the release criteria for license termination. In the case of dose modeling codes, the primary elements are:

- Mathematical models for the transport of radionuclides through the environment to a specific receptor, and
- The input parameters used in these models.

In evaluating compliance with the ALARA provisions and the radiological criteria for license termination, dose modeling codes serve as analytical tools to calculate the appropriate doses. These codes are normally designed to allow flexibility in the amount of input information required such that each licensee can optimize their dose analyses within the scope of license termination.

Dose modeling codes may contain either deterministic or probabilistic models, or both. These two types of models are classified by the format of the input parameters and the results (Reference 37). Deterministic dose modeling codes require the input of a fixed value (constant) for each input parameter and, in turn, produce a fixed output result (e.g. a DCGL value). It is recognized, however, that there is uncertainty inherent in the input parameter values and in the processes that affect the resultant dose. One way to address this uncertainty is to perform sensitivity analyses on key parameters using the deterministic model. Another method to address uncertainty is to model a range of values for key input parameters. This type of analysis is classified as probabilistic modeling. The result of a probabilistic analysis is a distribution of possible doses with some method to select the dose used for compliance being considered (refer to Section 3.6).

In performing dose assessments, a licensee will use available information such as the historical site assessment, site characterization data, plans for remediation, and/or final survey data. This information is processed in such a way as to determine the expected residual radioactivity present at the time of site release and to develop a source term model for the site. Then, an exposure scenario is established to guide the dose modeling parameter choice and lead to a reasonable assessment of the expected dose to the average member of the critical population group. The exposure scenario is chosen based on the potential future use of the site (refer to Section 3.2).

In the collection and integration of all these approaches, along with a general knowledge of environmental transport routes for the various exposure pathways, a conceptual model is developed that takes all of the preceding into consideration. The conceptual model itself is a qualitative depiction of the exposure pathways, the environmental transfer components, and how they all interact. With this model established, a series of mathematical models can be employed which essentially quantify the above processes by varying input parameters and/or boundary conditions, simplifying processes, and making specific assumptions. These mathematical models are typically packaged together in various computer codes (Reference 8).

The next two sections contain examples of dose modeling codes, which employ both of these approaches.

3.3.1 DandD Version 2.0

The Decontamination and Decommissioning (DandD) dose modeling codes implement and apply the scenarios, exposure pathways, models, assumptions, and justifications for input parameter choice, as outlined in NUREG/CR-5512, Volume 1 (Reference 38). The DandD code was developed by the NRC to compute doses over a thousand-year period and to report the results as an annual dose to a member of the critical population group. The possible exposure scenarios address residual radioactive contamination in soils and on building materials.

DandD 1.0 contains only a deterministic dose model. In the development of this model, the scenarios, assumptions, and default input parameters were defined to provide a reasonably conservative result. The parameters were also determined to be those values which best represented the expected variability across locations within the United States, in addition to the inherent variability among individuals (References 37 and 39). The justification for this approach was to provide both an effective means for performing screening analyses and to provide a simplified model by which input parameters can be tailored to site-specific data to perform site-specific dose assessments.

Version 2.0, includes the original deterministic dose model from Version 1.0, along with an added probabilistic module to support dose assessments with an accompanying uncertainty analysis. In the probabilistic version, the critical input parameters are no longer limited to only constant values, but now include the ability to enter distributions to account for variability or uncertainty in the default or site-specific values. The development of these distributions arises from variability across sites and the guidance provided in NUREG/CR-5512, Volume 3 (Reference 40). DandD, Version 2.0 can run with default input parameters distributions, which are built into the code, or modifications can be made to these distributions by inputting site-specific data. To perform a default analysis, the code only requires the input of the radionuclide concentrations. In using the default input parameters, along with a site source term, generic dose assessment is performed in which the results are unlikely to be exceeded and therefore represents a “reasonably conservative” determination (Reference 37, Section 1.3). When site-specific values are input to the code, certain exposure pathways may be eliminated and parameter uncertainty may be reduced (Reference 39). Any input of site-specific values causes the code to move to site-specific analysis. The computer code then determines and displays the results as dose distributions.

3.3.2 RESRAD

3.3.2.1 RESRAD Version 6.0

The computer code RESRAD (Residual Radioactivity) was developed by Argonne National Laboratory for the U.S. Department of Energy to guide environmental cleanup activities at radioactively contaminated sites. RESRAD is a pathway analysis computer code that calculates instantaneous radiation dose rates and excess cancer risks to a critical population group and derives cleanup criteria for radioactively contaminated soils, such as through the determination of DCGLs. The guidelines are established as allowable concentrations of residual radionuclides in soil, as well as in materials such as rubble or debris.

RESRAD can perform a variety of functions (Reference 41), including:

- Calculation of potential doses or lifetime risks to workers or members of the public from residual soil contamination.
- Calculation of the soil guidelines, such as DCGLs.
- Calculation of radionuclide concentrations in a defined medium (e.g. concentration in vegetation based on an initial soil concentration).
- Calculation of Area Factors that may be used to determine $DCGL_{EMC}$

The computer code accomplishes these tasks by pathway analyses (Reference 41), which include:

- Direct exposure from contaminated soil
- Inhalation of airborne radionuclides (includes radon progeny), and
- Ingestion of vegetation (growing in contaminated soil and water), meat and dairy products (from animals ingesting contaminated vegetation and water), fish (living in contaminated water), contaminated water, and contaminated soil.

The NRC sponsored a revision to the RESRAD code (Version 6.0), which included a probabilistic module capable of performing uncertainty analyses. As with most dose modeling codes, doses are derived through the use of models, which simplify complex systems and interactions. This simplification, along with uncertainty associated with model assumptions and future interpolation, results in output values, which contain an uncertainty component. Like its predecessors, this version of RESRAD incorporates default parameters based on a data set of national averages for deterministic parameters (Reference 42, Attachment A), as well as default parameter distributions (Reference 42, Attachment C) for performing probabilistic analysis.

The scenarios, parameters, and associated assumptions are intended to represent conservatively realistic cases and similarly provide conservatively realistic outputs. These guideline values, such as Derived Concentration Guideline Levels (DCGLs), are calculated through the use of an analytical approach called the concentration factor method (Reference 41, Section 1). In this approach, a relationship is established between initial radionuclide soil concentrations and the dose received by a member of the critical group. The relationship is reflected through a sum of the products of the pathway factors. These factors link compartments within the overall model and allow the results of the transfer of radioactive material from one compartment to another, or the emission of radiation from a compartment to be analyzed (Reference 43).

3.3.2.2 RESRAD-BUILD 3.0

RESRAD-BUILD 3.0 is a computer code from the RESRAD family of codes, which analyzes radiological doses and potential health impacts from remediation and occupancy of buildings contaminated with radioactive materials (Reference 42). The code is a pathway analysis model, which uses a combination of up to 3 compartments and several applicable exposure pathways to determine the appropriate guideline values. This analysis is structured to determine the potential dose to a member of the critical group from a building contaminated with residual radioactive material. The computer code incorporates the following features (Reference 44, Section 2):

- Pathways that include external exposure, inhalation of dust and radon, and ingestion of soil/dust.
- Modeling of up to 10 sources and 10 receptors for a building with as many as 3 compartments.
- Transport of radioactive material from one compartment of a building to another, calculated with an indoor air quality model.
- Multiple source geometries (e.g. point, line, area, and volume).
- Both surface and volumetric contamination
- Computation of the attenuation due to building materials (the model allows the choice of up to 8 material types each with an individual thickness and density).
- Multiple exposure scenarios (e.g., office worker, decontamination worker and building renovation worker).
- Determination of Area Factors for buildings that may be used in the determination of the $DCGL_{EMC}$.

RESRAD-BUILD is designed in a similar format to the RESRAD code, in that scenarios are constructed by the adjustment of the input parameters. RESRAD-BUILD also contains the same radionuclide listing contained in RESRAD.

As with RESRAD 6.0, RESRAD-BUILD 3.0 possesses a probabilistic module, which allows for the input of parameter distributions in order to perform uncertainty analyses (Reference 45). This probabilistic feature includes default data distributions and template files for non-radionuclide-dependent variables. This feature also allows the user to obtain results with both the peak-of-the-means and the mean-of-the-peaks method (refer to Section 3.6 for a discussion of these output methods).

3.4 Screening Analysis

Screening dose analyses may be performed with little site-specific information. Default parameters for a resident farmer and a building occupant are provided in both the DandD and RESRAD computer codes. These default parameters cover the movement of the radionuclides in the environment and the behavioral and metabolic parameters associated with the exposed person. The only site-specific information that is required is the source term characterization (the relative distribution of radionuclides). The default parameters built into the computer codes are intended to provide a reasonably conservative value for the DCGL determination. A screening analysis is appropriate for any site, providing that the modeling assumptions built into the computer codes is valid relative to the location and distribution of the contamination. For example, for soil contamination, the radioactivity is assumed to be in the top 6 inches (15 cm) of soil with no contamination in the groundwater (Reference 8, Appendix C).

Screening DCGLs for selected beta and gamma emitters are provided in Appendix C of Reference 8.

3.5 Site Specific Analysis

If compliance with the site release criteria discussed in Section 3.1 cannot be demonstrated with screening analysis, the licensee may move to a more realistic site-specific analysis. In the site-specific analysis, one or more of the default input parameters or dose pathways are replaced with site-specific data. Justification will be required for the site specific input data used. It should be noted that establishing technically acceptable site-specific parameters could require considerable resources to develop and ultimately gain approval of the involved stakeholders. Additionally, in some recent submittals, the use of site specific parameters were avoided due to concerns over their impact on the overall LTP approval schedule. Use of site-specific analysis may be justified by the need to establish DCGLs higher than the screening values due to the level of conservatism in the screening models.

An example of a site-specific analysis is discussed below. In this example, site specific values for the soil to plant uptake parameters were used to update the default distribution. The site-specific values were determined from knowledge of on-site soil types and literature data on soil to plant transfer. RESRAD 6.0 was then executed with the updated distribution and the results were presented for Cs-137 and Sr-90. The DCGL value for Sr-90 increased by a factor of 3 based on the use of the new distribution for the soil to plant uptake factor. The Cs-137 DCGL increased by approximately 33% based on the new distribution. The significant exposure pathway for Cesium-137 is direct exposure from gamma radiation. This pathway is not sensitive to soil to plant uptake factors. However, the dose from Strontium-90 is dominated by the ingestion pathway, which is highly sensitive to the updated parameter.

The above example points out the importance of knowing the sensitivity of the dose to a specific input parameter before investing resources in determining site-specific values for that parameter. The significant pathway for most of the gamma emitters (Cs-137, in this example) is direct exposure and only dose sensitive parameters relate to that pathway (e.g. shielding factor and occupancy factor). For the direct exposure pathway (gamma emitters), input parameters relating to the translocation of the contamination in the environment (e.g. soil to plant uptake factor) will have little dose sensitivity. Methods and results of sensitivity studies for both DandD and RESRAD are reported in References 43 and 46.

3.6 Use of Probabilistic Dose Models

Dose analysis is required in nuclear power plant decommissioning to determine if residual radionuclide-specific contamination levels would result in a dose that complies with the regulatory limit of 25 mrem per year (0.25 mSv/yr). EPRI published a report in 1999 entitled "Comparisons of Decommissioning Dose Modeling Codes for Nuclear Power Plant Use: RESRAD and DandD" (Reference 46), to assist the utilities in performing deterministic dose analysis. That report compared the two major dose analysis computer codes, DandD and RESRAD, by identifying and comparing important exposure pathways and capabilities as well as conservatism in the models, key parameters, and default input values.

Subsequent to Reference 46 being published, EPRI funded a study, which focused on the use of the RESRAD 6.0 computer code for probabilistic analysis of contaminated soil. The RESRAD 6.0 code incorporated the capability to perform Monte Carlo and Latin Hypercube Sampling (LHS) parameter uncertainty analysis, as well as maintaining the existing deterministic version of the RESRAD code entitled "Use of Probabilistic Methods in Nuclear Power Plant Decommissioning Dose Analysis" (Reference 58). The following three tasks were covered in this later EPRI work:

- Use of screening probabilistic dose analysis.
- Use of site-specific probabilistic dose analysis.
- General understanding of probabilistic dose analysis and its interpretation.

This section provides a summary of the latest EPRI study as guidance in performing probabilistic dose modeling.

Computer analyses were performed using a typical source term for a nuclear power station. The computer code was run in a screening mode using both the probabilistic and deterministic capabilities for the analysis of soil. Screening analysis means that the existing default inputs of the RESRAD 6.0 code were used except for the user-provided site-specific source term information. The source term radionuclides were determined from characterization results at several decommissioning nuclear power stations. The full suite of radionuclides included ^{3}H, ^{14}C, ^{55}Fe, ^{60}Co, ^{90}Sr, ^{63}Ni, ^{99}Tc, ^{129}I, ^{134}Cs, ^{137}Cs, ^{144}Ce, ^{238}Pu, ^{239}Pu, ^{241}Pu, ^{241}Am, ^{242}Cm, and $^{243}Cm/^{244}Cm$.

The results included the distribution of peak annual doses to the member of a critical group. The doses were determined for a unit soil contamination level (1 pCi/g) (37 Bq/kg) for each respective nuclide. These results are shown in Figures 3-1 and 3-2 for ^{137}Cs and ^{90}Sr. These figures show the distribution of doses determined by the probabilistic analysis as well as the point value determined by the deterministic analysis. For ^{137}Cs the dose range is approximately 1 order of magnitude with the mean dose being very close to the deterministic value. For ^{90}Sr the dose range is over 2 orders of magnitude showing much greater uncertainty in the results. For all of the radionuclides analyzed, with the exception of ^{129}I, the peak dose occurred during the first year. For ^{129}I, the peak occurred during year 3.

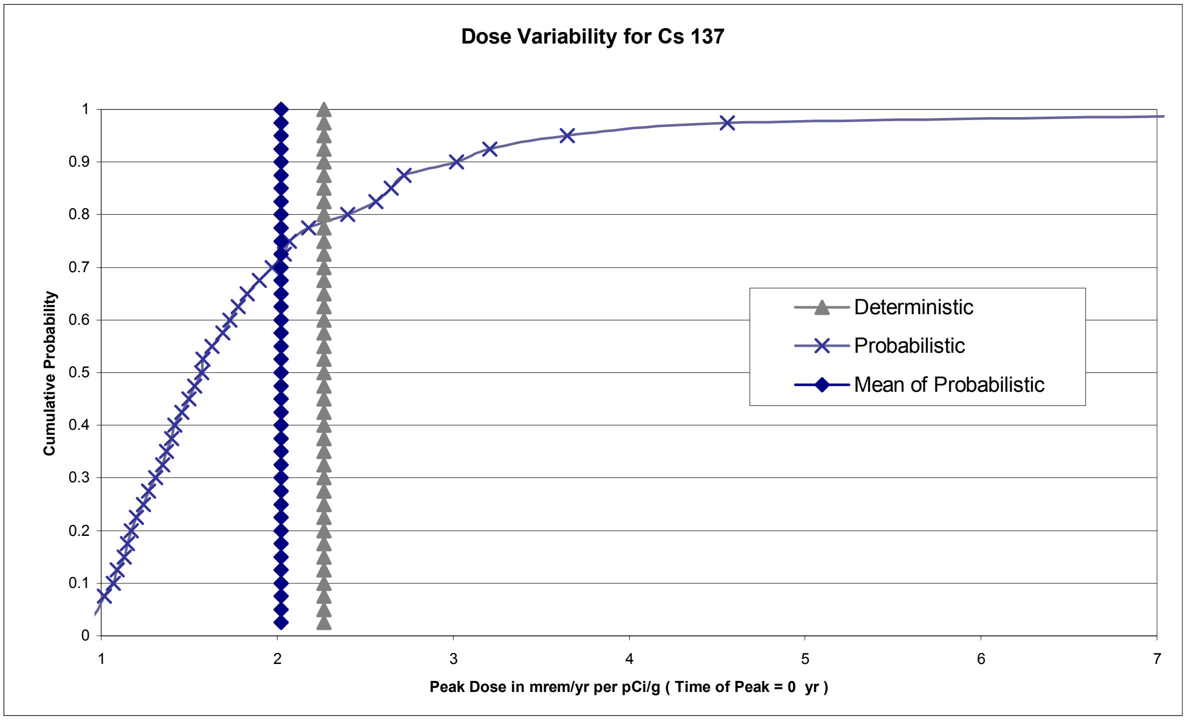

Figure 3-1
Comparison of Peak Dose between Deterministic and Probabilistic RESRAD Dose Analysis (Cs-137)

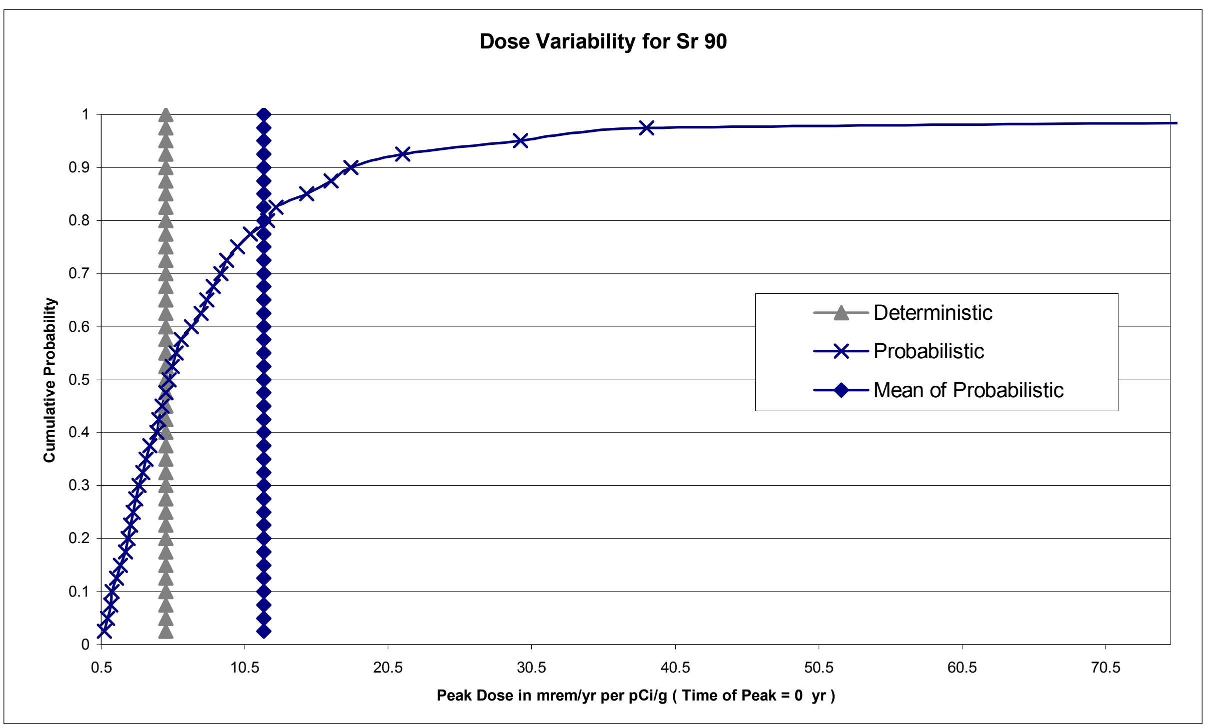

Figure 3-2
Comparison of Peak Dose between Deterministic and Probabilistic RESRAD Dose Analysis (Sr-90)

A summary of the screening results is presented in Table 3-1. This table shows the DCGL values determined by the probabilistic analysis as well as the deterministically calculated DCGLs and NRC screening values. The DCGL is the concentration of residual radioactivity distinguishable from background that, if distributed uniformly throughout a survey unit, would result in a total effective dose equivalent (TEDE) of 25 mrem per year (0.25 mSv/yr) to an average member of the critical group. The DCGL values derived from the deterministic analysis were higher than the NRC screening values except for ^{60}Co, ^{134}Cs, and ^{137}Cs. The DCGL values derived based on the mean of the peak dose from the probabilistic analysis were always higher than the NRC screening values. For ^{60}Co, ^{63}Ni, ^{90}Sr, ^{99}Tc, ^{134}Cs, and ^{137}Cs, the DCGL values from the probabilistic analyses were slightly higher than the NRC screening values. For ^{3}H, ^{14}C, ^{55}Fe, ^{129}I, ^{238}Pu, ^{239}Pu, ^{241}Pu, ^{241}Am and ^{243}Cm, the screening DCGLs from the NRC were much more conservative than the results from the probabilistic RESRAD analysis. Overall the results confirm the conservatism of the NRC screening values and indicate that probabilistic dose analysis can be effective in reducing conservatism in a DCGL derivation.

Table 3-1
Comparison of Screening DCGLs (NRC Screening Approach versus Results Using RESRAD Probabilistic Dose Analysis)

Nuclide	NRC surface soil screening values (pCi/g)	DCGL - Concentration (pCi/g) equivalent to 25 mrem/y for a specific value of Pcrit (Using Probabilistic RESRAD 6.0)				DCGL from deterministic analysis (RESRAD 6.0) (pCi/g)
	From NRC	Based on the mean of peak	Pcrit=0.5	Pcrit=0.10	Pcrit=0.05	Deterministic
H-3	1.1E+2	1.55E+3	1.68E+3	1.05E+3	9.48E+2	1.75E+3
C-14	1.2E+1	3.64E+1	4.01E+1	2.40E+1	2.10E+1	2.19E+1
Fe-55	1.0E+4	5.73E+4	7.60E+4	3.60E+4	3.10E+4	9.73E+4
Co-60	3.8E+0	4.40E-0	5.20E+0	2.80E+0	2.50E+0	2.82E+0
Ni-63	2.1E+3	2.83E+3	4.58E+3	1.51E+3	9.92E+2	5.45E+3
Sr-90	1.7E+0	2.11E-0	4.79E+0	1.40E+0	8.42E-1	5.01E+0
Tc-99	1.9E+1	2.12E+1	3.76E+1	1.18E+1	7.51E+0	5.36E+1
I-129	5.0E-1	8.93E-0	1.91E+1	5.94E+0	3.90E+0	3.85E+1
Cs-134	5.7E+0	6.63E-0	7.53E+0	4.55E+0	3.85E+0	5.01E+0
Cs-137	1.1E+1	1.24E+1	1.59E+1	8.28E+0	6.85E+0	1.10E+1
Ce-144	N/A*	3.19E+2	3.77E+2	2.21E+2	1.85E+2	2.03E+2
Pu-238	2.5E+0	4.73E+1	7.49E+1	2.51E+1	1.71E+1	6.31E+1
Pu-239	2.3E+0	3.68E+1	6.70E+1	2.08E+1	1.47E+1	5.69E+1
Pu-241	6.2E+1	2.19E+3	3.38E+3	1.19E+3	8.04E+2	3.01E+3
Am-241	2.1E+0	3.82E+1	6.10E+1	1.76E+1	1.45E+1	5.30E+1
Cm-243	3.2E+0	3.91E+1	5.39E+1	2.53E+1	1.98E+1	3.95E+1
Cm-244	N/A*	7.02E+1	1.22E+2	3.94E+1	3.11E+1	1.03E+2

* Not included in the NRC's list of screening DCGLs

To provide guidance in the performance of site-specific probabilistic dose analyses, several key input parameters were identified and example computer analyses were completed using site-specific inputs for these parameters.

Key parameters were identified from a probabilistic sensitivity analysis that involved running simulations, in which selected inputs are assigned distributions, while all other inputs are set to their default central value. The results of these analyses showed that site-specific key parameters included soil-to-plant transfer factor, thickness of unsaturated zone, K_d in the contaminated zone, density of the unsaturated zone, and, contaminated zone total porosity. Among these, the soil-to-plant transfer factor was the most significant for site-specific investigations.

Example analyses were performed with site-specific soil-to-plant transfer data and two key radionuclides of concern in nuclear power plant decommissioning, i.e., ^{137}Cs and ^{90}Sr. The input distributions for site-specific soil-to-plant transfer factors for ^{137}Cs and ^{90}Sr were derived based on a combination of literature data using soil conditions for a site selected as a test case and the default input distributions, and use of the Bayesian technique. The resulting differences in DCGLs between the screening probabilistic analysis and site-specific analysis are summarized in Table 3-2. The NRC screening values shown in Table 3-2 were obtained from Reference 8 and determined through the use of the DandD computer dose modeling code. The DCGL values for several critical points, e.g., 50%, 90%, and 95% confidence levels (corresponding to Pcri=0.5, 0.1, and 0.05), are listed in this Table. The results show that the use of site-specific data can lead to a significantly higher DCGL for a given site. In this case the DCGL value for ^{137}Cs increased approximately 50% over the NRC screening value. The DCGL for ^{90}Sr, however, increased by a factor of 5 based on the site specific input values for the soil-to-plant transfer factor.

Table 3-2
Comparison of Soil DCGLs (pCi/g) calculated with Screening and Site-Specific Probabilistic Dose Analysis

	Cs-137		Sr-90	
	Screening	Site-specific	Screening	Site-specific
Pcrit = 0.5	15.2	18.0	4.83	10.7
Pcrit = 0.1	8.36	11.6	1.35	5.21
Pcrit = 0.05	6.36	10.3	0.95	4.28
Based on the peak of the mean	12.5	16.3	2.62	8.71
Based on the mean of the peak	12.5	16.3	2.62	8.71
NRC screening value	11		1.7	

Table 3-2 provides both the peak of the mean and the mean of the peak DCGL values. The peak of the mean dose represents the maximum value in the mean dose curve that is composed of the mean dose values from all simulations at each time step. The mean of the peak dose represents the mean value of the peak dose calculated for each simulation in the entire simulation time period. The table shows that the peak of the mean values for both radionuclides is equal to the mean of the peak values. This was also the result for all radionuclides analyzed, providing that the peak dose value occurs during the first year. As mentioned above, ^{129}I was the only radionuclide analyzed for which the peak dose occurred in a later year (year 3). For ^{129}I, the peak of the mean and the mean of the peak doses differed by less than 3%. The NRC suggests using the peak of the mean as the appropriate criterion (Reference 8, Appendix C), however it does not appear that there is a significant difference between the two measures with the use of RESRAD 6.0.

4
SITE REMEDIATION PLANS

Plans for site remediation are a required component of the License Termination Plan (LTP), in accordance with 10 CFR 50.82 (a)(9)(ii)(C), "Termination of License". The following provisions must be addressed in the site remediation section of the LTP:

- The dose from residual radioactive contamination to the average member of the critical group must not exceed the total effective dose equivalent (TEDE) of 25 mrem/yr (0.25 mSv/yr).
- The dose to the public from residual radioactive contamination has been reduced to levels that are as low as reasonably achievable (ALARA).

The successful performance of remediation activities and ALARA evaluations requires that several parameters be taken into consideration:

- The extent of contamination: surface area, volume, concentration
- The depth of contamination
- The type of the surface: walls, ceilings, rough and painted surfaces
- Decontamination factor for specific remediation techniques
- Consumables
- Labor resources: number of workers, work time
- Support tasks: preparation, transport, waste processing, maintenance
- Waste generation
- Airborne contamination potential

This section of the LTP will summarize remediation actions, methods, ALARA evaluations, and the criteria that will be employed to demonstrate compliance. Detailed discussions on each of these subjects are provided below.

4.1 Remediation Actions

In remediation planning, it is important to understand the various remediation actions that may be undertaken in the remainder of the decommissioning process. Such actions are typically associated with specific media types and can be categorized under one of the following (Reference 47, Section 4.3 and Reference 48, Section 4.4):

- Land Areas: Soils

- Structures: Interior and exterior building surfaces, major exterior
- structures, paved exterior ground surfaces, and plant system exterior surfaces
- Non-structural Plant Systems: Interior surfaces of process piping and components

Discussions pertaining to remediation should focus not only on the methods used in the remediation, but also on the techniques and procedures used to ascertain the effectiveness of such activities. This includes the use of dose modeling codes as discussed in Chapter 3. In the planning and preparation of remediation actions, industry experience such as the results from previous decontamination activities, should be accessed to increase efficiency and maximize benefits.

Methods that may be employed to reduce the levels of residual radioactivity below the applicable DCGL or ALARA value will be described in the following sections. In addition to these methods, action levels will be discussed that will initiate specific remediation procedures.

4.1.1 Soils

As discussed in Chapter 2, soils will be surveyed in accordance with the site characterization program to determine the location, depth, and magnitude of soil contamination. Soil, concrete rubble (demolished structures), asphalt rubble, and other soil-like materials will be analyzed to determine the presence and quantity of residual radioactive contamination. If the residual radioactivity is determined to be above the site release criteria of 10 CFR 20.1402, the affected soil will be remediated. Remediation involves the removal of soil, gravel, asphalt, and other soil-like materials, as necessary to meet these criteria.

Remediation action levels for surface soils are typically determined by the applicable average surface activity $DCGL_W$ (the DCGL used for the statistical tests, see Reference 1, Section 2.5.1.2) and the ALARA evaluation (Section 4.3). Areas of elevated residual radioactivity above the $DCGL_{EMC}$ (the DCGL used for the elevated measurement comparison (EMC), see Reference 1, Sections 2.5.3 and 2.5.4) would also serve as an action level to initiate remediation for the affected area. Thus, an entire survey unit or any portion thereof may be remediated if the applicable action level is exceeded. Definition and discussion of these DCGLs is provided in greater detail in Section 2.3.2 of this report.

Soil and all similar materials to be remediated will be removed with appropriate excavation equipment, with care being exercised to prevent the spread of contamination and minimize the generation of airborne contaminants. This contaminated soil material will be disposed of as low-level waste.

The majority of the potential site soil contamination will be associated with locations inside the restricted areas. The remediation of such areas should be performed following the removal of structures, components, or systems that might provide interference, such as interior floors, tanks, etc. Remediation of soils may also be required outside of the radiation control area due to past site activities. Remediation in these areas will usually not be impeded by buildings or structures.

Remediation surveys should be performed to ensure that changing soil contamination profiles are assessed, such as setting up a field lab for gamma spectroscopy to assist in the assessment of soil excavation results.

4.1.2 Structures

Structures are typically surveyed for residual contamination once interfering plant systems and components have been removed. If the structural material is determined to be clean, then the material can be released as scrap or disposed of in an appropriate manner. If the structural material is found to be contaminated above the applicable DCGL or ALARA levels, then remediation actions must be performed so that the material meets the radiological release criteria for either a standing structure or for rubblized concrete. Removal of certain structures may be necessary, as structural integrity may have become compromised by remediation activities. It may also prove cost effective to remove structural materials as radioactive waste, thus eliminating iterative ALARA decisions and Final Status Survey of these materials.

Remediation actions performed on structural materials vary according to the level of residual radioactive contamination present. A number of factors should be considered when choosing an appropriate remediation method for a given application, including:

- The size of the contaminated area
- The degree of contamination
- The fixed/removable fractions
- The surface material being remediated
- The depth of the residual contamination
- The level of accessibility to the contaminated area

A variety of remediation techniques exist for structures. These can be classified as being one of two basic approaches, non-invasive and invasive removal of contamination. The methods discussed are intended as examples, and not as a comprehensive review of techniques.

The first approach combines methods that employ cleaning techniques to remove residual contaminants on or near the surface. For both metal and concrete surfaces, various surface-cleaning methods can be used, which include wet or dry wiping, vacuum collection of dirt and contamination, and high or low-pressure washing. Washing and wiping can be effective methods for situations that include (Reference 49, Section 4.2.2):

- Remediation of stairs and rails.
- Remediation of structural surfaces, metals, and materials.
- Structural areas, which are difficult to access and deny the use of other decontamination methods.
- Surfaces that require a cleaning reagent.

Wet and dry wiping are labor-intensive methods that are normally conducted during remediation activities on structural surfaces. Wet wiping constitutes the use of wipes with a decontamination reagent, while dry wiping encompasses the use of wipes and an alternative cleaning matrix, such as an oil-impregnated media. Effective decontamination reagents can include a variety of industrial cleaning solutions, as well as certain household cleaning compounds. However, it should be anticipated that household cleaning solutions will generate large wastewater volumes that must subsequently be processed. This type of remediation technique is most effective for the removal of loose surface contaminants. However, it should be understood that for such a high level of effectiveness, the cost of labor and other resources can become high. Consideration should be given to those factors that can affect the effectiveness of decontamination. These include: the level of effort used, the type of reagent or "binding" media used with the wipe, and the physical and chemical characteristics of the residual contamination being removed.

Pressure washing is a method that removes contaminants that have become embedded into the surfaces of structural media, such as deposits of oil, grease, and boron. The equipment includes the use of a directional nozzle and an attached wet vacuum to collect the contaminated water (containing the contaminants forced from the structural material). Such units typically clean at approximately the same rate as dry vacuum systems and can involve the use of decontamination reagents. The effectiveness of the pressure washing method is based on the following factors: the cleaning reagents used, the media being remediated, the physical and chemical properties of the contaminant, and the water pressure.

In some cases, structural material, such as concrete, may contain activation products or contamination to a depth inaccessible by the above cleaning techniques. In such instances, a second approach to remediation must be considered which encompasses mechanical and/or physical removal of the activated/contaminated media from the structure. Removal techniques include scabbling, abrasive blasting, grinding, core drilling, and manual cutting and sectioning of the affected area. In choosing the most appropriate method for remediation, consideration should be given to removal depth, control of airborne contamination, and the minimization of generated waste volumes. In addition, the minimization of personnel exposure and the logistics of waste packaging and disposal should be included in the selection process. Outlined below is a list of removal techniques that can be employed in the invasive remediation of structures:

- Scabbling: A routine decontamination technique that removes contaminated structural surface material (i.e. surface concrete) by a mechanical pounding action. Scabbling equipment may employ rotopeen hammers, flappers, bush heads, or similar devices. The action of the devices removes surface deposits and residual contamination that is located close to the surface.

- Concrete Planing: A decontamination technology that removes contaminated structural surface material (i.e. surface concrete) by shaving a predetermined amount of concrete from structures, coupled with a vacuum extraction system that collects the generated waste material (Reference 50).

- Centrifugal Shot Blasting: A decontamination technology that removes contaminated structural material by using hardened steel shot propelled at high speed. The depth of decontamination can be controlled by both the volume and speed of the shot. The shot is reused (until repetitive use pulverizes the shot into waste material), while the removed structural material is collected by means of vacuum collection system (Reference 50).

- Abrasive Blasting: This method is an effective removal technique for structure surfaces that may not be necessarily smooth (i.e. embedded piping, drains, and other structure penetrations). This technique uses a blast media which is either recyclable (best for minimization of waste volume) or is lost in the generated waste (i.e. grit blasting). The abraded materials are then captured by a vacuum system. Abrasive blasting can also be employed for materials other than concrete such as steel and other building materials.
- Manual Removal/Sectioning: For structural materials that are volumetrically contaminated, the affected section can be cut out with abrasive cutting instruments, or diamond wire saws, and disposed of as low level radioactive waste. This method may be used for removal of concrete down to the first layer of rebar. Devices such as chipping guns and jackhammers can also be used for more aggressive removal and can often be coupled with vacuum capture systems to collect the loosened material (Reference 51, Section 4.3.1).
- Core Boring/Rock Splitting: This technique involves the localized removal of structural material, such as concrete. This method generates low noise/vibration, has a controlled rate of removal with a variable cutting speed, and the associated water-cooling minimizes airborne particulate suspension. The drawbacks to this method are that it cannot penetrate rebar and the used cooling water has to be recycled or collected as waste material (Reference 52, Section F.1.3).
- Needle Gunning: This remediation method is intended for the removal of surface contamination in areas that are too difficult to access (i.e. inside corners and cracks) by other removal techniques. The system uses a pneumatically operated device with thin steel rods to abrade a concrete surface, while an attached vacuum system captures the removed material.
- Strippable Coatings: Coatings can be applied in order to stabilize or remove loose contamination present on surfaces where more aggressive remediation techniques are not appropriate.

In addition to these remediation techniques, demolition can also be considered if the area is volumetrically contaminated, such as walls or shield blocks that have been subject to neutron activation.

For further information regarding remediation techniques for concrete and hazardous waste materials, refer to the EPRI Reports, "Concrete Decontamination Technology Workshop Proceedings" and "Hazardous Waste Material Remediation Technology Workshop" (References 50 and 53, respectively).

4.1.3 Systems

Contaminated non-structural plant systems and components are commonly removed for processing either on site or off site for disposal. If the final material is radioactive it is disposed of as low-level radioactive waste. If the contamination is removable, the systems and components may be decontaminated and released. Typical decontamination methods are those that employ chemical means as discussed in Section 4.1.2.

Embedded piping can be handled in different ways depending on the location and extent of contamination. Embedded piping located close to the surface can be removed with appropriate methods during demolition of the surrounding structure. If the structures are not to be removed, consideration should be given as to whether the pipe can be left in place, and if so, is remediation of the pipe interior surface required. These decisions will depend on the level and type of contamination in the pipe and on the DCGL determined from dose modeling. Mechanical and chemical decontamination by one of the methods discussed above may be possible. The shielding effectiveness of the material surrounding the pipe may be factored into the dose analysis. The embedded piping can also be sealed and grouted to mitigate potential doses from liquid or airborne pathways. For additional information on embedded pipe remediation practices, refer to the EPRI Report, "Embedded Pipe Decontamination Technology Workshop" (Reference 54).

Remediation methods for other systems and components typically involve wet/dry wiping, high and low-pressure washing, abrasive blasting, scabbling, spalling, chemical decontamination, and/or removal and disposal of the affected portion(s).

4.2 ALARA Evaluations

Subpart E of 10 CFR Part 20 requires that the dose to an average member of the critical population group from residual radioactivity at a site released for unrestricted use, not exceed 25 mrem/yr (0.25 mSv/yr). In addition, Section 20.1402 requires that the residual radioactivity be reduced to levels that are as low as reasonably achievable (ALARA). The 25 mrem/yr (0.25 mSv/yr), therefore, becomes a ceiling that represents the upper limit of dose allowed for unrestricted use. An ALARA evaluation will determine if the remediation goal should be lower than this ceiling.

4.2.1 ALARA Screening Levels

Screening ALARA evaluations should be performed for each type of media that will remain at the time of license termination. Various remediation methods are discussed above. Each of the methods chosen for remediation should be included in an ALARA analysis. For soil, only excavation need be considered as an effective means of remediation. The ALARA evaluation for soil will determine whether excavation is necessary and, if it is, what ALARA concentration is required. For building surfaces, there may be several remediation techniques under consideration (e.g. wiping and scabbling). Each of these would have an ALARA evaluation to determine which one, if any, is justified.

The screening ALARA evaluation is performed under the assumption that normal conditions will exist. That is, the cost evaluation should not assume that any unusual difficulty will occur during remediation. This will produce a conservative result that may be generically used for the site. This analysis need only be performed once for each remediation method being considered. The results of this ALARA analysis will determine which methods must be employed for remediation if the survey unit already meets the established $DCGL_W$ value. It will also determine the concentration level below which no remediation will be required.

4.2.2 Area-Specific ALARA Evaluation

The screening evaluation discussed above will be applicable for most of the areas and media being considered for remediation. However, selected areas may present special challenges that would require an area specific ALARA evaluation. This area specific evaluation would result primarily from remediation costs that would be significantly higher than was assumed in the screening evaluation. Building surfaces that are not readily accessible or contaminated soils beneath building floors would be examples of areas that may require a specific ALARA evaluation due to high remediation costs. These specific evaluations would result in an ALARA radionuclide concentration level that is higher than the ALARA screening value (determined above) and may demonstrate that no remediation of these areas is necessary.

4.2.3 Methodology

The ALARA analysis is intended to be an unbiased (realistic) evaluation of remediation actions, to avoid doses to members of the public and unnecessary costs. The ALARA level, that results from the analysis is the level that balances the avoided dose with the costs. That is, the dollar value of the avoided dose is equal to the total cost of the remediation. At the time of this report the value of a person-rem avoided may be taken as $2000 (Reference 7).

The avoided dose is simply the dose savings of the remediation times the number of people that could be exposed. The dose savings is the dose rate from residual radioactivity times the fraction of the contamination that would be removed by the remediation times the number of years that exposure could occur. The avoided dose should be corrected for radioactive decay and for the present worth of future dose. The radioactive decay may be accounted for by using the average life of the radionuclide ($1/\lambda$) in place of the number of years of exposure. Ignoring the present worth will result in a conservative answer, i.e. a greater likelihood of remediation being required.

Remediation costs include the cost of the: a) remediation action, b) radioactive waste packaging and handling, c) waste shipping, d) waste burial, e) dose received by workers during remediation and waste handling, and f) traffic fatalities during waste transport. Normally the last two costs can be ignored without significant impact on the result of the analysis.

The NRC has developed equations and provided acceptable parameter values for performing an ALARA analysis in Appendix D of Reference 8. Using methodology and parameter values in this Reference, the licensee need only determine the total cost of remediation and the fraction of the contamination that will be removed by the remediation to complete the analysis. The results of the analysis in Reference 8 are expressed in terms of the concentration of residual radioactivity that would be considered to be ALARA. This value may be greater that the $DCGL_w$ (remediation must then meet the $DCGL_w$) or less than the $DCGL_w$ (remediation must be attempted if the actual concentration exceeds the ALARA concentration).

4.2.4 Industry Experience

In addition to the regulatory and guidance resources cited above, there is valuable information, which can be gathered from the License Termination Plans of nuclear power plants currently in the submittal phase. From these LTPs, useful information can be drawn upon, such as that outlined in ALARA cost estimate data and remediation effectiveness evaluations. In the following sections, these industry experience points are summarized to facilitate the understanding of the remediation component of an LTP.

A more detailed synopsis of the remediation data from these LTPs is addressed in a parallel EPRI report, which will be published in July of 2002. This report, titled "Summary of License Termination Plan Submittals by Three Nuclear Power Plants", Reference 59, provides a summary the content of LTPs submitted to the NRC by Connecticut Yankee, Maine Yankee, and Trojan.

4.2.4.1 Estimated Costs for ALARA Evaluation

Cost estimates for soil remediation (removal) are in the $2,000-$4,000 per cubic meter range (Note that costs can vary based on site-specific conditions). This includes digging, packaging, transport, and disposal, with the disposal cost being the largest contribution. With costs in this range it is unlikely that the ALARA analysis would ever indicate that remediation of soil is required below the $DCGL_w$ level. The NRC acknowledges this, even for a remediation cost of $667 per cubic meter, in Reference 8 (Appendix D, Section 1.4).

Cost estimates for concrete scabbling range from approximately $90 to $150 per square meter. These estimates include the labor and equipment costs for the scabbling and the disposal cost for the concrete removed. The wide range of cost estimates is due, in part, to various depths of concrete removed and the accessibility of the areas being remediated. In a sample ALARA analysis the NRC used a scabbling cost of $50 per square meter (Reference 8, Appendix D). This analysis resulted in the ALARA screening level being approximately equal to the $DCGL_w$ (ALARA concentration = 0.97 $DCGL_w$). Depending on the site specific input parameters used in the analysis, it is possible that scabbling will result in a cost benefit for surface concentrations less than the $DCGL_w$.

4.2.4.2 Evaluation of the Effectiveness for Remediation Methodologies

A useful tool in the performance of remediation for various needs is the effectiveness of the available methodologies. By assessing this effectiveness, the user can select the most appropriate approach for specific site conditions.

Remediation effectiveness can be determined by assessing the fraction of the residual radioactive contamination that is removed for a given remediation method. This determination can be made by collecting and analyzing a sufficient number of pre-remediation and post-remediation measurements to establish a consistent fraction of the removed material (Reference 48, Section 4.2.3).

In the absence of such a determination, remediation effectiveness can be estimated from other decommissioning nuclear power plants and studies performed. Listed below are remediation methods, which have been assessed for their effectiveness at one decommissioning nuclear power plant site (Reference 49, Section 4.4.1):

- **Scabbling:** The scabbling equipment has a production rate of 20 ft^2/hr (1.8 m^2/hr) at a depth of ¼ to ½ of an inch (0.64 to 1.3 cm). The effectiveness of this system is assumed to be 95%, which is considered reasonable, as structural contamination is assumed to be located within or close to the surface.
- **Pressure Water Washing:** This decontamination system has a cleaning rate of approximately 100 ft^2/hr (9.3 m^2/hr) and generates about 5.4 liters of liquid per square meter. The effectiveness of this system is dependent on the media being washed, the composition of the contaminants, the water pressure, and the cleaning reagent(s) used. In the case of loose contaminants, the effectiveness can be high. However, when remediating hard-to-remove contamination, the effectiveness can be assumed to be approximately 25%. By using reagents and a slower cleaning speed, the effectiveness can be increased, but can result in higher costs.
- **Wet and Dry Wiping:** This decontamination method has an estimated cleaning rate of 30.1 ft^2/hr (2.8 m^2/hr) and a waste generation rate of 5 liters per hour (82.8 cm^3/hr) for the wet method. The effectiveness of this system is dependent on factors, which include the reagent(s) used, the necessary level of effort, and the composition of the contaminant. The effectiveness of this system is assumed to be 100% for loose contaminants, whereas the hard to remove component is assumed to have a reduction efficiency of approximately 20%.

Various concrete decontamination technologies were also assessed in terms of their capabilities, cleaning rates, and costs through the DOE concrete decontamination test program (Reference 50). Shown below are the results for some of the assessed technologies:

- **Scabbling:** This concrete decontamination method comes in various forms and has an average scabbling rate of 33 ft^2/hr (3.1 m^2/hr) at a depth range of ½ to 1 inch (1.27 to 2.54 cm) (the work area is dependent on the bit size used). Operation of the equipment varies in its labor requirements, depending on whether the system is handheld or mobile, and requires the support of an air compressor/HEPA filter vacuum system. The technology is beneficial in terms of its proven capabilities, integrated dust removal system, and its ability to decontaminate concrete in corner areas. The system is limited in terms of its ergonomics and maintenance, and requires that the surface being decontaminated is dry. The cost of this system varies with the scabbling unit used.

  An example of one such scabbling system is a unit called the Squirrel-III Floor Scabbler. This system was assessed to have a production rate of 30 ft^2/hr (2.8 m^2/hr) at a depth of 1/8 of an inch (0.32 cm), and generated waste material at a rate of 1 drum per 1,500 ft^2 (139.4 m^2). The system operates at a flow rate of 60 ft^3/min (1.7 m^3/hr) at 90 psi. This system is beneficial in its minimization of airborne particulate generation with a 100% collection efficiency for dust and debris.
- **Marcrist DTF25 Floor Shaver:** This concrete decontamination system is a specific type of concrete planer (see Section 4.1.2) that shaves a pre-determined amount of concrete from a structure floor. This self-propelled system uses a rotating drum, lined with diamond blades and collects waste material by means of a dust shroud/vacuum extraction system. This concrete planer has a production rate of 50 ft^2/hr (4.6 m^2/hr) at a depth of 0.35 inches (0.89 cm), with a removal width of 10 inches (25.4 cm) and a removal gap of 6 inches (15.2 cm). Operation of the equipment requires 1 operator and 1 hose tender. This system generates a fine concrete powder, and so a dust collector filter is necessary to minimize airborne

particulates. Thus, the waste elements generated from its operation are the removed structural material, spent dust collection filters, and spent drums (after removal of 20,000 ft^2 (1,858 m^2) at a depth of ¼ of an inch (0.64 cm)). This system carries an estimated cost of $32,000, with replacement drums carrying an added cost of $10,000 each.

- **Corner-Cutter Needle Gun:** This concrete decontamination system is a specific type of needle gun (see Section 4.1.2). This system has a production rate of 20 ft^2/hr (1.8 m^2/hr) at a depth of 1/8 of an inch (0.32 cm) with a waste generation rate of 1 drum per 1,500 ft^2. This technology is beneficial in terms of its minimization of airborne contamination and ability to decontaminate ceiling and corner areas.
- **Centrifugal Shot Blasting:** This concrete decontamination method typically has a production rate of 173 ft^2/hr (16.1 m^2/hr), a depth range of ½ to 1 inch (1 to 2 cm) and a removal width of 15 inches (38 cm). Operation of the equipment requires 1 operator and 1 technician and an air compressor/HEPA filter vacuum system. The technology is beneficial in terms of its aggressive cleaning performance and minimization of secondary waste. The limitations of this system are in the areas of maneuverability, maintenance, creation of projectile hazards, and that the surface being decontaminated must be dry and even. This system carries an estimated cost of $150,000.

A specific system called the Centrifugal Shot Blaster was assessed to have a production rate of 292 ft^2/hr (27.1 m^2/hr) at a depth of ¾ of an inch (1.9 cm). Operation of the equipment requires 1 operator and 1 assistant. This self-propelled system generates a fine concrete powder, thus a dust collector is integrated into the system to minimize airborne particulates. The waste elements generated from its operation are the removed structural material, spent dust collection filters, and spent steel shot (shot eventually becomes pulverized into a dust and is collected with the removed structural material). This system carries an estimated cost of $150,000.

- **Ultra High Pressure Water:** This concrete decontamination method has a production rate of 42 ft^2/hr (3.9 m^2/hr) at a depth of ¼ of an inch (0.64 cm), with the work area being dependent on the bit size used. Operation of the equipment requires 1 operator and 1 technician, along with the support of an air compressor system. This technology is beneficial in terms of its minimization of airborne contamination, ergonomics, and ability to decontaminate concrete in corner areas. The limitations of this system are that of water processing, short effective range, and the final waste form. This system carries an estimated cost of $160,000.

5
FINAL STATUS SURVEY PLAN

The final phase of license termination is the Final Status Survey (FSS), which is performed to demonstrate compliance with dose-based regulations. Chapter 3 of this document describes the process of developing a site-specific dose-based limit. The FSS represents verification that the calculated limits have been satisfied.

The guidance document for developing an FSS Plan is the Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM, Reference 1). MARSSIM methodology is based on the Data Quality Objectives (DQO) process to ensure that the final status survey results are of sufficient quantity and quality to support the final decision.

This chapter provides a simplified summary of the major considerations in planning and developing a site-specific FSS program. The following are focused on providing insight gathered from field experience and provide the framework of the FSS plan. Detailed technical information that may be readily obtained from guidance documents, such as MARSSIM, is discussed in general terms.

5.1 Preliminary Survey Considerations

The initial development phase of a site-specific FSS establishes the framework for selecting the appropriate survey design. In order to complete this phase, a thorough evaluation of the site, which can include contaminants, area-specific contamination potential, materials of interest, and other site-specific parameters, must be evaluated. This evaluation expands upon the information gathered during the historical site assessment, and facilitates the process of delineating the site into distinct areas based on specific characteristics.

5.1.1 Select Method for Determination of Background

One of the more challenging aspects of FSS design is the method of evaluating site-specific background, especially for structural surfaces. Because certain radionuclides may occur at significant levels as part of background in the media of interest, most sites typically do not possess the necessary conditions for a simplified survey design where the contaminant is not present in background.

MARSSIM presents a method to select background reference areas, based on similar characteristics. A calculated number of measurements/samples are then collected, based on the selection of the statistical test. Because of the difficulty of identifying a non-impacted structure with similar radiological characteristics (e.g., concrete surfaces greatly vary in the content of natural uranium and associated progeny), a more recent trend has been to utilize the survey unit itself to develop an appropriate background value. Two examples of alternate methods are presented in the EPRI report "Considerations for Determining Background Radiation Levels in Support of Decommissioning Nuclear Power Facilities" (Reference 24).

5.1.1.1 Background Reference Areas

Background reference areas are selected based on similar physical, chemical, geological, radiological, and biological characteristics (Reference 1, Section 4.5). The reference area may be non-impacted (no reasonable potential for residual radioactive contamination), or may be associated with the survey unit being evaluated, given that it is not potentially contaminated by site activities.

The non-impacted status of the background reference area is assessed through historical information and/or the collection of survey/sample data. This selection of these areas is typically of great interest to site regulators and stakeholders and can represent an in-depth and time-consuming evaluation. Typically, a site will develop a technical basis document that discusses the selection process as well as the intended use of the data. The following factors should be considered:

- Site radiological releases that may have affected the area.
- Ambient concentrations of man-made radionuclides that did not originate from the site (e.g., Pu-239 and Cs-137 from atmospheric dispersion of fallout from nuclear weapons testing).
- Year of construction (in relation to site impacted structures).
- Geographical variations (e.g., elevation variation, structural background variability, variation in soil composition, and fallout deposition in soils).
- Variations in material types.

The lack of available non-impacted areas may prompt the use of impacted areas for background determinations. For this case, the user might designate a small area for performing isotope-specific measurements (e.g., *in-situ* gamma spectroscopy) to confirm the absence of site contaminants of concern. Another option involves a statistical evaluation of the gross results from the survey unit data, which may include verifying the "fit" of the data to a specific distribution to assess whether the data set represents a background distribution. An evaluation of outliers to the data set can also provide information on potential residual radioactive contamination locations. However, it is important for the user to realize that such non-traditional methods of background determination may be more difficult to justify, simply given the limited application of these methods. Accordingly, a sound technical basis and inclusion of the site stakeholders in the development are critical.

The user must also be aware that the initial selection of background reference areas will typically prove to be inadequate to represent the entire site (especially relevant to large facilities). Thus, the technical basis document should allow the user flexibility to perform future background determinations.

5.1.1.2 Statistical Background Determination

A calculated number of measurements/samples are collected, based on the statistical test selected, and the data set is compared to the data set collected in the survey unit (Wilcoxon Rank Sum Test). A simplified variation of this method is to calculate the mean of the background data set, subtract that mean value from each survey measurement/sample result, and use the Sign Test (Reference 55). A more detailed discussion of survey design is provided in Section 5.3. Alternate options for background determination are discussed in the previous section.

5.1.2 Identify Contaminants and Establish DCGLs

A primary objective of scoping and characterization surveys is to identify the site-specific contaminants of concern. Contaminants are normally identified for each media type through samples that are subject to isotopic analysis. Derived Concentration Guideline Levels (DCGLs) are then developed for each radionuclide (see Chapter 3).

Three methods can be used in comparing survey results with the DCGL(s). The first is a direct statistical comparison of survey/sample results with the DCGL(s). This method is most commonly used for sites with a single contaminant of concern, or for soil sampling (when isotopic analysis is performed).

The second method involves the use of surrogate measurements to demonstrate compliance with the DCGLs for all contaminants of concern (Reference 1, Section 4.3.2). Due to the nature of nuclear power reactor facilities, this method will typically be applied. The objective of the surrogate method is to simplify the survey/analysis process and reduce costs by accounting for a certain percentage of "hard-to-detect" radionuclides in quantifying results from the survey/analysis method. This is performed by establishing a surrogate ratio (the ratio of hard-to-detect radionuclides to detectable radionuclides) for the selected analysis method. The ratio can then be used to develop a modified DCGL for the selected analysis method. An example of a common situation where the application of this method may be prudent is to account for alpha-emitting nuclides when performing beta field measurements. The presence of tritium, iron-55, or carbon-14 may also represent a scenario where the use of the surrogate method is desirable.

The equation for calculating a modified DCGL is as follows (Reference 1, Equation 4-1):

$$DCGL_{DET,\mathrm{mod}} = DCGL_{DET} \times \frac{DCGL_{HTD}}{\left[\left(C_{HTD} / C_{DET}\right) \times DCGL_{DET}\right] + DCGL_{HTD}} \quad (4)$$

where:

$DCGL_{DET,mod}$	=	modified DCGL
$DCGL_{DET}$	=	DCGL for detectable radionuclide
$DCGL_{HTD}$	=	DCGL for hard-to-detect radionuclide
C_{HTD}	=	concentration of hard-to-detect radionuclide
C_{DET}	=	concentration of detectable radionuclide

A third method of comparing the results against DCGLs may be employed in situations where multiple radionuclides must be considered. Under such conditions the DCGLs are modified to account for these multiple radionuclides. The surrogate method discussed above is one such way of adjusting DCGLs to account for cases of both singular and multiple hard-to-detect radionuclides. Other approaches include the use of the unity rule and developing a gross activity DCGL (for surface activity).

The unity rule is used to assess a survey unit when multiple radionuclides are present. The equation shown below is used by summing the fractional contribution of each radionuclide to its DCGL. If the resulting total is less than or equal to 1, then the radionuclide mixture meets the dose criteria set for the survey area (Reference 1, Equation 4-3):

$$\frac{C_1}{DCGL_1} + \frac{C_2}{DCGL_2} + \frac{C_3}{DCGL_3} + \ldots \frac{C_N}{DCGL_N} \leq 1$$

Where:

C	= Concentration of each radionuclide
DCGL	= Derived Concentration Guideline Level for each radionuclide
N	= Number of radionuclides

Another approach where multiple radionuclides are present as surface contamination, is to develop a gross activity DCGL. This method is beneficial in that it allows the surveyor to perform field measurements of gross activity for comparison to the DCGL. The gross DCGL is developed by first determining the relative fraction of the total activity, f, for each radionuclide. Once this is done, the fractions are divided by their respective DCGLs and summed. The inverse of this total then yields the gross activity DCGL as shown below (Reference 1, Equation 4-4):

$$DCGL_{gross} = \frac{1}{\frac{f_1}{DCGL_1} + \frac{f_2}{DCGL_2} + \frac{f_3}{DCGL_3} + \ldots \frac{f_N}{DCGL_N}}$$

Where:

f	= relative fraction of each radionuclide to the total activity
DCGL	= Derived Concentration Guideline Level for each radionuclide
N	= the number of radionuclides

The user should be cognizant that the above equation may not be applicable for sites where the radionuclides considered have highly variable concentrations relative to each other. In such situations, the user may want to select the most conservative surface contamination DCGL from the list of radionuclides considered (Reference 1, Section 4.3.4).

5.1.3 Classify Areas

The frequency of survey/sampling in a given area will vary based on contamination potential. For this reason, areas must be classified prior to the performance of a FSS. The information required to assess contamination potential and to classify an area is typically collected during the historical site assessment, scoping, and characterization phases. All data/information are evaluated.

Classification may be performed for a given area (e.g., the entire containment structure), or on a survey unit level. The definitions of given classes are as follows (Reference 1, Section 4.4):

- Non-Impacted Areas: Areas that have no reasonable potential for residual contamination.
- Class 1 Areas: Areas that have or had, prior to remediation, a potential for radioactive contamination or known contamination (e.g., the containment and auxiliary building).
- 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 $DCGL_W$ (e.g., BWR turbine building).
- 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 $DCGL_W$ (e.g., PWR turbine building, provided there were no primary to secondary leaks).

The $DCGL_W$ is a specific concentration guideline level intended to be used for statistical tests and is derived from the basis of an average concentration over a large surface area (Reference 1, Section 2.5.1.2).

5.1.4 Identify Survey Units

A survey unit is a physical area for which a separate decision will be made as to whether or not the area exceeds the release criterion (Reference 1, Section 4.6). The site may be divided into survey units at any time prior to the FSS. A survey unit must comprise an area of the same classification and is typically limited to the following size (based on guidance in MARSSIM):

Table 5-1
Guidance for developing survey unit size limits

Survey Unit Classification	Structures	Land Areas
Class 1	100 m^2 floor area	2000 m^2
Class 2	100 to 1000 m^2	2000 to 10,000 m^2
Class 3	No limit	No limit

MARSSIM Section 4.6
Note the above recommended areas differ from the RESRAD default values of 10,000 m^2 for land and 36 m^2 for structures.
[Actual size limits may vary for the site of interest]

Note that, in some cases, it may be prudent to exceed the established size limits (e.g., a single Class 1 room may consist of 2,150 ft^2 (200 m^2) floor area, but it is desirable to have a single survey unit for that room). For this situation, the number of required measurements can be corrected to account for the increased size. For example, given the survey unit size limit of 1,080 ft^2 (100 m^2), an actual room area of 2,150 ft^2 (200 m^2), and a calculated required number of measurements of 15 per survey unit, the number of measurements is thus multiplied by two to correct for the increased survey unit size.

5.1.5 Select Instruments and Survey Techniques

The FSS will typically include the use of a wide range of instrument types in order to optimize data quality for each measurement type. For additional discussion on the numerous factors that must be considered in the selection of appropriate instruments and survey methods refer to Section 5.4.1. Following the selection of the appropriate instrument, a program should be developed to support its use. A discussion of the minimum recommended requirements for an instrument program is provided in the following sections.

The selection of instrumentation and survey techniques is interrelated, given that the survey technique is dependent solely on the instrument selected, and vice versa. The user may choose to select an instrument based on practical usability in the field. In contrast, the selection of an instrument simply based on the technical capabilities may result in implementation difficulties. Thus, both issues should be considered in the DQO process.

The level of detail provided in the License Termination Plan (LTP) regarding the instrumentation selected for the site will be determined based on negotiations between the site and the regulators. The user must consider the fact that a period of months or years could pass prior to implementing the FSS Plan presented in the LTP, during which time enhanced technology will be developed. Thus, the technical specification requirements of the instrumentation may be more appropriate for inclusion in the LTP versus the listing of specific instruments.

5.1.5.1 Selection of Instruments

The primary considerations in the selection of survey instruments include radiation(s) of concern, minimum detectable concentration (MDC), which can be calculated or derived empirically, and field reliability. Note that due to the complexity of the MDC concept, calculations should be performed during the development of the License Termination Plan (LTP) in order to assure all interested parties understand and agree to the selected detection system(s).

When selecting field instrumentation, it is important to consider source efficiency, which until recently has not been a parameter considered when calculating the total efficiency of the survey method. Source efficiency is defined as the ratio of the number of particles of a given type emerging from the front face of a source and the number of particles of the same type created or released within the source per unit time (Reference 1, Section 6.5.4). For an ideal source, the value of source efficiency is 0.5 (recommended for beta emitters with endpoint energies > 0.4 MeV). However, when considering typical media types found in the field (e.g., concrete, cinderblock, wood, etc.), the actual value may be much less.

The consideration of source efficiency is most important when measuring alpha or low energy beta emitters, given the increased probability for attenuation in site media. The recommended source efficiency for alpha and low energy beta emitters (endpoint energies of 0.15 to 0.4 MeV) is 0.25. A site may also choose to empirically derive the actual source efficiency given their contaminant(s) of concern and field conditions.

The total efficiency, as presented in MARSSIM, is the product of the instrument efficiency ε_I (defined as the ratio of the net count rate and the surface emission rate of a source for a specified geometry), and the source efficiency ε_s. Note that the surface emission rate is the number of particles of a given type above a given energy emerging from the front face of the source per unit time (2π particle fluence). Traditional calibration methods consider the total activity (4π emission) of the source when calculating detection sensitivity. Thus, it is important to ensure that calibration source vendors provide certified values for surface emission rates. Note that while MARSSIM presents a specific method for instrument calibration, other industry-accepted calibration methods may be selected based on site-specific factors.

A description of typical instrumentation used for field measurements and their applications is presented in Tables 6.1 and 6.2 of MARSSIM.

5.1.5.1.1 Calibration and Maintenance

Calibration refers to the determination and adjustment of instrument response in a particular radiation field of known intensity (Reference 1, Section 6.5.4). For most sites, the calibration requirements for the FSS will not differ from those applicable for routine operations (i.e., job coverage surveys). Some of the primary differences that should be considered for the FSS are described below.

- The applicability of source efficiency (particularly for alpha-emitting nuclides) (refer to Section 5.3.2.1)
- The effect of environmental conditions on instrument response (especially applicable for outdoor surveys)
- Source-to-detector geometry for field use (should be consistent with calibration)
- Mechanical and thermal stresses on the instrument (given the day-to-day use of the instruments)

The minimum recommended frequency for calibration is annually (Reference 1, Section 6.5.4), or following maintenance that might affect response. Calibration can be performed in accordance with an industry-recognized standard, or per vendor technical specification requirements for a particular instrument.

5.1.5.1.2 Response Checks

Consistent with any radiological survey, a program for performance testing instrumentation should be developed for the FSS program. Guidance for response tests, such as frequency, source strength, acceptable range, etc., can be obtained from industry-recognized standards or vendor recommendations. The selected criteria should be consistent with the DQOs. For most field instruments, the acceptable range of response is ± 20% (Reference 1, Section 6.5.4).

Factors that should be taken into consideration to enhance a routine performance test program are 1) the requirement to perform a post-survey performance test and 2) the utilization of the performance test data to evaluate trends in the instrument. The first recommendation allows the user to provide evidence that the instrument was operating properly before and after data collection (i.e., a problem was not experienced during the course of survey that may have affected instrument response). The second allows for in-process review of instrument performance to determine if a bias may exist in the data. A scatter-plot of the daily performance tests can be a useful indicator of downward or upward trends in instrument response (refer to Section 5.3.1.3.2).

5.1.5.1.3 MDC Calculations

The determination of the MDC is discussed in MARSSIM, primarily because the survey design is highly dependent on the sensitivity of the selected instrumentation. Due to this fact, users of this document should be cognizant that traditional industry methods of MDC determination may not provide an adequate technical basis for the cited MDCs. Also, the NRC staff has identified deficiencies in reporting the characterization results with respect to MDC determination. For example, in some cases maximum and average surface beta contamination results were below the stated MDCs. Reporting the results in such a way leads to an unclear depiction of whether or not the MDCs are representative of an entire survey unit or of multiple areas within it. Characterization survey results are an important element of the license termination process, thus they should be presented in a manner that is technically defensible (Reference 2).

MARSSIM provides a discussion and several formulas for the calculation of *a priori* direct and scan measurement MDCs for various radiations of concern (Reference 1, Section 6.7). The *a priori* MDC is calculated prior to the performance of the survey and describes the detection capability of the instrument. Because the MDC is calculated *a priori*, conservative estimates of the efficiency and background should be used in the calculation. MARSSIM recommends that a measurement system with an *a priori* MDC between 10-50% of the DCGL be selected (does not apply to scan MDCs), while acknowledging that this goal may not be achievable (based on cost restrictions, etc.). It is important to recognize that this recommendation is provided as an administrative control to assure that overly optimistic MDCs are not reported. Thus, if a conservative estimate of the MDC is performed, which would include the consideration of the measured background and minimum efficiency for the detection system, an MDC less than or equal to the DCGL is acceptable. This further supports the argument that the consideration of source efficiency, which may decrease the overall detection system efficiency and increase the value of MDC, is an important factor.

It is also important to recognize that the calculation of MDC for a modern detection system, such as a large-area position sensitive proportional detector, may deviate from the methods described in MARSSIM. In addition, the collection of empirical data from field testing (with sources of known activity) can provide a defensible estimate of the MDC. The objective is to provide a technical justification for the detection system sensitivity that satisfies the DQOs.

5.1.5.1.4 Measurement Uncertainty

The uncertainty associated with measurement results for a particular detection system should be reported with each individual result (Reference 1, Section 6.8). Measurement uncertainties are described as either systematic or random. Systematic uncertainties arise from the application of fixed parameters in calculating measurement results that are consistently higher or lower than the true value. An example is assuming a single efficiency value for a detection system, when the efficiency is actually lower.

Random uncertainties refer to fluctuations associated with a known distribution of values (Reference 1, Section 6.8.1).

The practical approach to classifying uncertainty is to, 1) design the survey to minimize measurement uncertainties and 2) determine the appropriate quantity to be reported (to quantify the uncertainty). MARSSIM recommends several practices that can reduce the uncertainty for a given measurement system, including the following (Reference 1, Section 6.8.1):

- Select a detector that will minimize the potential uncertainty - Use a detector that has a stable response and is not sensitive to expected environmental conditions or conditions of use
- Apply the appropriate efficiency value for the surface being measured (Refer to discussion of source efficiency in Section 5.1.5.1)
- Use standard measurement procedures/protocols
- Ensure instrument operators are trained and experienced
- Perform quality review of data (Refer to Section 5.3.1.3.2)

The parameter that is typically selected to express uncertainty for field and laboratory data is the standard deviation (σ). Section 6.8.3 of MARSSIM provides a summary of the calculation of standard deviation.

5.1.5.2 Selection of Survey Techniques

The survey technique(s) for a specific site is selected based on type and distribution of contaminants, type of survey (e.g., scan versus static), field use, cost, and use of the data (as determined per the DQO process). Table 4-1 of MARSSIM provides a list of typical instrumentation types for specific contaminants and recommended survey methods. A detailed discussion of survey methods is provided in Section 5.3.

5.1.5.3 Considerations for Selection of Sample Collection and Direct Measurement Methods (Open-Land Areas)

The FSS of open-land areas presents an entirely new set of challenges, particularly for sites concerned with hard-to-detect and subsurface contamination. Typically, the release of open land areas is achieved with a combination of sampling and surface scanning. However, surface scanning may not be practical for subsurface and/or hard-to-detect contaminants (such as alpha-emitters). In addition, the presence of subsurface contamination necessitates the development of a depth-distribution profile of the contaminants, which is required to establish appropriate survey methods and data analysis. The DQO process, based on site characteristics, will provide the necessary information to select the appropriate survey/sample method.

5.1.6 Site Preparation

The tasks included in site preparation are receiving consent to survey, establishing the property boundary, evaluating the physical characteristics of the site, establishing a reference coordinate system, and developing survey unit maps (Reference 1, Section 4.8). It is assumed that the first two tasks have been achieved prior to the initiation of the development of the FSS plan. A discussion of the remaining three tasks is provided in the following sections.

5.1.6.1 Physical Characteristics of Site

The physical characteristics can serve as a gauge for the difficulty of the survey. For instance, a site represented by a few small structures that have been completely stripped of equipment and components will require a simple survey design. In contrast, a large site with multiple structures, embedded pipe, and numerous open land areas will require a more complex and multi-faceted survey design. The degree of difficulty in accomplishing the FSS may prompt the site to reconsider decommissioning strategies, such as removing non-load bearing walls and wall penetrations prior to commencing FSS. Given the potential impact to cost and scheduling, the importance of evaluating the characteristics of the site becomes clear.

5.1.6.2 Reference Coordinate System

A reference coordinate system may be developed for the entire site or for an individual survey unit. The coordinate system is represented by a grid of “x” and “y” coordinates, which divides the area into squares of equal area. The grid is referenced to a single location or benchmark such that random and systematic measurements can be located. Note that the benchmark should be a fixed point of location that can be used to relocate measurements at any time following the completion of the survey.

5.1.6.3 Maps

Following the designation of specific survey units, maps can be developed. The maps must be to scale (given the use of the reference coordinate system), and should account for the entire surface area of the survey unit. Structures are typically depicted as “folded out,” meaning the walls, floors, and ceilings are diagramed as though they lay along the same vertical plane. The reference coordinate system overlays the map to allow for identification of measurement locations. Structural components, such as beams and columns, should also be accounted for, particularly for Class 1 and Class 2 survey units (given limits on survey unit size and the higher potential for contamination). Open land areas are simple to depict, except when large variations in topography exist.

Note that the development of scale maps can be a time consuming process, and requires personnel with knowledge of drawing programs such as AutoCAD® or TurboCAD®. Therefore, it is prudent to begin map development as soon as possible and preferably prior to the commencement of FSS, in order to avoid schedule delays.

5.2 Survey Design and DQO Process

The appropriate survey design for a given site is developed during the DQO process, which consists of seven steps (Reference 1, Appendix D):

1. State the problem
2. Identify the decision
3. Identify inputs to the decision
4. Define the study boundaries
5. Develop a decision rule
6. Specify limits on decision error
7. Optimize the design for obtaining data

While this process may appear at first glance to be cumbersome, the recognition of the necessary outputs from the process serves as a simplification. Appendix D of MARSSIM breaks down each step of the DQO and the expected outputs for each step. A summary of the minimum expected outputs and a brief discussion of each is as follows:

1) Classify and specify survey unit boundaries.
   - *Identify and classify survey units based on contamination potential (a survey unit is typically represented by a room or area that has common boundaries and is easily distinguished from other survey units).*
2) State the null hypothesis.
   - *For a typical survey design, the null hypothesis (H_0) is: the residual radioactivity in the survey unit exceeds the release criterion.*
3) Specify a gray region.
   - *upper bound of gray region is the $DCGL_{W.}$*
   - *lower bound of the gray region (LBGR) may be set at the mean concentration of the survey unit, or may be initially selected as one half the $DCGL_W$ (adjusted to provide an acceptable value for relative shift).*

4) Define Type I and Type II decision errors and their associated probability limits.

   - *Type I (α) Error, typically 0.05 (although referred to as a "false positive" error, alpha actually represents a false negative conclusion in relation to the presence of elevated activity in the survey unit).*

   - *Type II (β) Error, typically of less interest to regulators given that it represents the error of falsely accepting the null hypothesis that residual radioactivity exceeds the release criterion.*

5) Estimate the standard deviation of the measurements in the survey unit.

   - *Characterization data from the survey unit should be used to determine the standard deviation. MARSSIM suggests that if no data are available, the standard deviation may be represented by a coefficient of variation of 30% (additional explanation in Section 5.2.2).*

6) Specify the relative shift.

   - *An expression representing the resolution of the measurements in units of measurement uncertainty, represented by Δ/σ and generally designed to have a value between one and three.*

7) Specify the detection limits.

   - *Specific or minimum required MDCs for each measurement system or type.*

8) Calculate the estimated number of measurements and measurement locations.

   - *Given the selected statistical test, calculation is based on relative shift and Type I and Type II error rates (typically increased by 20% to account for uncertainty in the parameters used to calculate the estimated number of measurements).*

   - *Measurement locations needed for the statistical tests are random (Class 3) or systematic (Class 1 and 2).*

   - *Systematic and scanning measurements are performed for Class 1 survey units to provide assurance that small areas of elevated activity comply with the release criterion ($DCGL_{EMC}$).*

   - *Note that the same number of measurements would be required for Class 1, Class 2, and Class 3 survey units, given that the survey units meet appropriate size limits (Table 5-1).*

9) Specify the documentation requirements for the survey.

   - *Survey plans, survey unit data packages, summary reports, etc. as determined on a site-specific basis.*

It should be noted that software was developed to simplify the MARSSIM survey design process. This software, called the Computerization of MARSSIM for Planning and Assessing Site Surveys (COMPASS) is available to the public on the web (http:www.orau.gov/essap/marssim.htm#Compass).

5.2.1 Selecting the Appropriate Statistical Test

While the selection of the statistical test is presented as part of the Data Quality Assessment (DQA) in MARSSIM, the user should recognize the fact that, similar to the DQO process, the DQA process is iterative and is not necessarily performed in a linear sequence. Because the estimated number of measurements is calculated based on the statistical test selected, this determination is performed as part of the survey design.

MARSSIM recommends that a preliminary data review be performed prior to selecting the appropriate method of data analysis (i.e., parametric or non-parametric statistical tests). This review involves the calculation of simple statistical parameters (e.g., mean, median, standard deviation) and graphical analysis (e.g., histograms, scatter plots) to determine the appropriate use of the data. A simplified data design that would be appropriate for most sites involves selecting a non-parametric test during the survey design phase, given that no assumptions of data symmetry are assumed for these tests.

The two most commonly used non-parametric tests are the Wilcoxon Rank Sum Test (WRS) and the Sign Test. Given the fact that structural survey units commonly represent more than one material with varying natural background activity (when performing gross measurements for alpha and beta radiation), the Sign test is more user-friendly. MARSSIM states that the Sign test (one-sample statistical test) should only be used if the contaminant is not present in background and radionuclide-specific measurements are made. However, recent publications support the use of the Sign test when the contaminant is present in background (Reference 55). A more complete development of this concept can be found in Section 12.3 of Reference 56. Additional discussion is also provided in the following sections.

5.2.1.1 Wilcoxon Rank Sum Test

The WRS test is a two-sample test that compares the background reference area data set with the survey unit data set to accept or reject the null hypothesis. This test is most effective when residual radioactivity is uniformly distributed over the survey unit (Reference 1, Section 8.4.1) and when the contaminant is present in background.

The user should note that this test is only practical when the survey unit is comprised of materials similar to those present in the reference area, or is delineated based on material type.

Specific guidance on applying the test is presented in Section 8.4.2 of Reference 1.

5.2.1.2 Sign Test

The Sign Test is a one-sample test that is selected when the contaminant is not present in background or is present at a small fraction of the $DCGL_w$. Thus, a background reference area is not necessary and individual results are compared directly with the $DCGL_w$.

As discussed above, this test is typically selected when the contaminant is not present in background; however, recent publications support the use of the Sign test when the contaminant is present in background (Reference 55). For this use, the mean of the background data set is calculated. The calculated value is then subtracted from each survey unit or survey sample result. The Sign Test is thus performed for the net data values.

Specific guidance on applying the test is presented in Section 8.3.2 of Reference 1.

5.2.2 Calculating Parameters for the Statistical Test

The parameters that must be established during the Data Quality Objectives (DQO) process in order to perform the statistical test include the following:

- Alpha (α): Type I decision error or false positive error (null hypothesis is rejected when it is true).
- Beta (β): Type II decision error or false negative error (null hypothesis is accepted when it is false).
- **LBGR**: Lower Bound of Gray Region.
- Delta (Δ): shift or the width of the gray region ($DCGL_w$ – LBGR).
- Standard deviation **(σ)**: estimated standard deviation of the data set.
- Relative Shift **(Δ/σ)**
- Required number of measurements (**N)**

Note that the statistical test is set up such that the objective is to reject the null hypothesis (concluding that the median concentration of residual radioactivity in the survey unit is less than the $DCGL_w$). Accordingly, a Type I decision error (α) actually represents an "incorrect rejection error" result, meaning that the survey unit meets the release criteria when it truly does not. Thus the selected rate for a Type I error is of interest to regulators and stakeholders. The default value for α is typically 0.05.

The Type II error rate is selected by the site to optimize survey design, considering the acceptable rate of "incorrect non-rejection error" (falsely concluding the survey unit does not meet the release criteria, when it truly does) results, to ensure that the test has sufficient power (1-β) to detect residual radioactivity concentrations at the LBGR. Beta values typically range from 0.025 to 0.10.

The gray region is the range of values for a survey unit where the consequences of making a decision error are relatively minor. The upper bound of the gray region is always set at the $DCGL_w$, while the LBGR is typically set to the mean concentration in the survey unit, or, in the absence of data, may be set to one-half of the $DCGL_w$ (i.e., LBGR = 0.5 * $DCGL_w$). The shift (Δ) can then be calculated by subtracting the LBGR from the $DCGL_w$.

The estimated standard deviation of the data set should be determined with existing data. In the absence of data, or if the sensitivity of the survey instrument is too low to adequately define a standard deviation, it can be assumed to be 30% of the $DCGL_w$ (or a 30% coefficient of variation). The coefficient of variation is defined as σ/μ (or $\sigma/DCGL_w$). The relative shift is then calculated by Δ/σ.

The required number of measurements can then be calculated given the above parameters and the calculations provided in Section 5.5.2.2 of MARSSIM. However, the recommended method is to refer to look-up tables (Reference 1 Tables 5.3 and 5.5). Note that the values provided in these tables take into account a 20% increase in the actual calculated number of measurements. The purpose of this increase is to account for the uncertainties in the estimated values of the measurement variability, specifically the estimated survey unit standard deviation.

5.2.3 Determine Measurement Locations (Random or Biased)

Measurement locations are either random or systematic. Random locations are selected via random number generation. Systematic locations are evenly spaced, with the initial location selected via random number generation.

In order to locate random locations within a survey unit, a two-dimensional scale drawing is developed with a reference coordinate system (typically in one-meter increments) grid overlay. Note that three-dimensional surfaces must be represented in the drawing by folding out vertical surfaces (i.e., walls).

The spacing for systematic locations can be triangular or square. A square grid is recommended for simplicity. The spacing for each type is calculated as follows:

$$L = \sqrt{\frac{A}{N}} \quad \textit{for a square grid} \quad \text{(Reference 1, Equation 5-8)}$$

$$L = \sqrt{\frac{A}{0.866N}} \quad \textit{for a triangular grid} \quad \text{(Reference 1, Equation 5-7)}$$

Where:

L = spacing

A = total area of survey unit

N = required number of measurements

In accordance with MARSSIM, Class 3 survey unit measurement locations should be random in nature. Class 1 and Class 2 survey unit measurement locations should be systematic in nature.

5.2.4 Survey Investigation Action Levels

Investigation or Action Levels are incorporated into the survey design to determine when additional investigations may be necessary. Investigation levels also serve as a quality control check for the measurement process. Investigation level flags may indicate problems with survey design, the misclassification of a survey unit, a potential instrument problem, the presence of naturally-occurring interference, such as short-lived radon progeny, and other unexpected conditions.

When an investigation level is exceeded, the first step should be to confirm that the result actually exceeds the investigation level (for gross measurements). This typically involves the collection of additional measurements/samples. Depending on the result, the survey unit may be reclassified, remediated, or resurveyed.

The most acceptable method for determining investigation levels is using a statistical-based parameter. For example, a level might be arbitrarily established at the mean plus 3 standard deviations (Reference 1, Section 5.5.2.6). Another example might be to calculate the level at which 95% confidence is achieved in scanning (applicable for automated types of survey instrumentation).

The example provided in MARSSIM for investigation levels is presented below in Table 5-2. Note that these levels represent an example and may not be appropriate for all sites (especially for sites where very conservative or default DCGLs are selected, or where detector sensitivity is very close to the $DCGL_w$).

Table 5-2
Example Investigation Action Levels

Survey Unit Classification	Direct Measurement or Sample Result Flag	Scan Measurement Result Flag
Class 1	> $DCGL_{EMC}$ or > $DCGL_W$ and > a statistical parameter-based value	> $DCGL_{EMC}$
Class 2	> $DCGL_W$	> $DCGL_W$ or > MDC
Class 3	> fraction of $DCGL_w$	> $DCGL_W$ or > MDC

MARSSIM Table 5.8

When an investigation level is exceeded, there are several possibilities that should be evaluated, including reclassification, remediation, resurvey, and an evaluation of the survey design. The appropriate follow-up actions to a flagged result should be integrated into the survey design.

There are also specific scenarios that should be considered prior to the commencement of an investigation, including the following:

- Verify that the original result is accurate and representative of the radiological state of the area (i.e. the original result was not a "false-positive").

- Verify that the elevated result is due to residual radioactive contaminants.
- Investigate the origin of the contamination (i.e. historical operation/process or the result of D&D activities) in order to determine the appropriate follow-up actions.
- Evaluate if a pattern of contamination (i.e., isolated or easily delineated area) exists by reviewing other survey/sample results (especially applicable to media samples).

An example of the actions that may result when an investigation level is exceeded is provided in Table 5-3.

Table 5-3
Example Investigations for Class 3 Survey Unit

Condition	Follow-up Actions
Single total surface activity measurement in excess of 50% of $DCGL_w$	▪ Reclassify
Acceptance of null hypothesis (survey unit does not pass statistical test)	▪ Reclassify ▪ Evaluate need for remediation ▪ Evaluate survey design
Elevated scan result	▪ Quantify result (perform total surface activity measurement at flagged location) ▪ Delineate area of elevated activity ▪ Evaluate need for remediation ▪ Reclassify

5.3 Field Measurement Methods

Measurement is defined in MARSSIM as 1) the act of using a detector to determine the level or quantity of radioactivity on a surface or in a sample of material removed from a media being evaluated or 2) the quantity obtained by the act of measuring (Reference 1, Section 6.1).

Measurement is commonly referred to as survey or sampling. Two types of surveys can be performed, direct measurements and scanning. Direct measurements may be discussed as total surface activity, static, or fixed surveys, and refer to an integrated measurement collected over a given time interval while the detector is held in a stationary position. Scanning typically refers to a dynamic survey performed at a pre-determined rate, based on desired sensitivity or MDC. Sampling typically pertains to collecting a volumetric sample of the media of interest for laboratory analysis.

5.3.1 Survey Methods

The compliance with dose-based limits typically involves the measurement of total surface activity (alpha, beta, etc.) for structures and systems and the quantification of radionuclide concentrations in environmental media (soil, water, and air) of open land areas. In some instances, a volumetric limit for structural media (e.g., concrete) may also be desirable.

Several factors must be considered in selecting a measurement method, including the following (Reference 1, Section 6.2.1):

1) Type of measurement to be performed
2) Target detection limits (MDC) (refer to Section 5.1.5)
3) Radionuclide(s) of interest
4) Environmental conditions potentially affecting survey measurements
5) Field use and instrument durability
6) Scan surface area
7) Analytical precision and bias
8) Frequency of quality control measurements/samples
9) Cost of the method being evaluated
10) Necessary turnaround time
11) Specific background for the measurement type/radionuclide of interest
12) DCGLs
13) Measurement documentation requirements
14) Measurement tracking requirements

The consideration of these issues individually and collectively will provide the elements for establishing a successful survey/sample program. As an example, a health physicist might select a proportional detector in order to achieve the necessary detection limit. The instrument technician may select a sealed gas proportional detector instead for ease of field use (i.e., eliminating the need to tote a supply gas bottle). However, the impacts of field use and environmental conditions may result in the need to repeat surveys due to failing instrumentation.

Another example would be the surface area that will require scanning. For large sites, it may be prudent to consider the use of large area detectors to minimize the time required for scanning. Current technology offers automated large area detection systems that decrease total scan time by optimizing scan rate versus required detection limits and automatically generate data reports.

The use of data loggers is generally recommended to fulfill data quality requirements by minimizing transcription errors and providing an electronic record. Data loggers also tend to reduce the total survey time.

5.3.1.1 Scanning

The performance of scan surveys, which typically involve the movement of a portable radiation detector across a surface, is generally the most challenging of the field surveys. The scan is a most crucial component of the survey, given the assumption of the statistical design that the survey unit has relatively uniform levels of residual radioactivity. The scanning of large areas can compensate for areas where there is a low probability of detecting “hot spots” when performing direct measurements.

The specific objective of scanning is to identify localized areas of elevated residual radioactivity that may require additional investigation (e.g., direct measurements) or action. The detection sensitivity of scanning is typically set at a fraction of the $DCGL_{EMC}$.

The degree of difficulty for the scan survey is a function of the contamination of concern. While beta and photon emitting nuclides are generally easy to detect, scanning for alpha emitting nuclides can be difficult due to the overall efficiency of the measurements.

Another important factor to consider is the use of automated versus manual detection systems. The traditional manual detection system, where the surveyor utilizes a portable detector and flags elevated results based on visual and/or audible signals, may be preferred due to its simplicity and practicality. However, the use of an automated system that electronically records data may provide a higher degree of confidence in the result by eliminating human error and may eliminate the need to collect direct measurements (Reference 1, Section 5.5.3). Other factors to consider include detection capabilities, cost, schedule, field durability, data quality, and availability of resources. The selection of a particular scan instrument should be evaluated against the established DQOs in order to ensure an appropriate match for the site of interest.

5.3.1.2 Direct Measurements

Direct (total surface activity) measurements are collected by placing the detector over the area of interest for a specified interval and recording an integrated result. The random or systematic measurement locations, as calculated for the specific statistical test (refer to Section 5.2.2), are used. Biased direct measurements may also be performed in other areas that have a potential for elevated activity and/or at areas that are flagged during the scan survey. The direct measurement is generally simple to perform. In addition, the determination of the MDC for direct measurements is relatively simple compared to that for scanning.

5.3.1.3 Sampling

Sampling is typically performed for open land areas, given the difficulty of quantifying activity in the field for porous materials. The need for sampling may also arise for structures with painted surfaces or volumetric contamination. Sample locations can be random, systematic, or biased (analogous to direct measurements). Structures with painted surfaces that have the potential for contamination within the paint matrix may require both direct measurements and samples at the same location.

Qualified individuals trained in the use of the necessary procedures and equipment should perform the collection of samples. Thus, additional training may be required even for experienced survey technicians. Due to the nature of sampling, the quality control requirements (as established by the Quality Assurance Project Plan or QAPP) are typically more cumbersome than those for field data collection.

Sample analysis should only be performed by laboratories that have been evaluated against the FSS quality requirements. This will include a review of written procedures and protocols prior to establishing a contractual agreement with the lab.

5.3.1.3.1 Identify Data Needs

The requirements for the sample data are established during the planning phase of the FSS. Because the development and implementation of a sample collection plan can represent a difficult task, small-scale sites may choose to contract the entire sampling and analysis process to an experienced and qualified vendor. A list of sample data requirements to consider is as follows (Reference 1, Section 7.2.1):

- Type of samples to be collected/analysis to be performed
- Radionuclides of interest
- Number of samples to be collected
- Type and frequency of QC samples
- Amount of material required for each sample (based on analysis type)
- Standard operating protocols (SOPs) to be followed
- Analytical bias and precision
- Target detection limits
- Cost of analysis type
- Required turnaround time
- Sample preservation and shipping requirements
- Background of the radionuclide of interest
- DCGL for each radionuclide of interest
- Documentation requirements
- Sample tracking requirements

5.3.1.3.2 Data Quality Indicators

The Quality Assurance Project Plan (QAPP) must also include the requirements for data quality. The discussion of data quality will assist in the selection of the appropriate analytical method. Examples of the typical data quality indicators and their MARSSIM definitions (Reference 1, Section 7.2.2), as well as methods to evaluate the indicator, are described below. Note that the data quality indicators are not specific to sampling and apply to all survey methods. However, the applicability of each indicator is dependent on the type of survey.

1) Precision (quantitative)

   A measure of agreement among replicate measurements of the same property under prescribed similar conditions.
   - collocated samples
   - field replicates
   - analytical laboratory replicate
   - laboratory instrument replicate

2) Bias (quantitative)

   The systematic or persistent distortion of a measurement process that causes error in one direction.
   - reference material
   - performance evaluation samples
   - matrix spike samples
   - background samples
   - field blanks
   - method blanks

3) Representative (qualitative)

   A measure of the degree to which data accurately and precisely represent a characteristic of a population parameter at a sampling point.
   - measurement method
   - size of sample collected

4) Comparability (qualitative)

   A qualitative term that expresses the confidence that two data sets can contribute to a common analysis and interpolation.
   - utilization of same measurement system for all analyses

5) Completeness (quantitative)

   A measure of the amount of valid data obtained from the measurement system, expressed as a percentage of the number of valid measurements that should have been collected.

6) Other Data Quality Indicators

   - delineation and classification of survey units
   - decision error rates
   - variability in contaminant concentration (typically presented as the standard deviation)
   - lower bound of the gray region
   - detection limit

5.4 Data Quality Assessment and Interpretation of Survey Results

The Data Quality Assessment (DQA) process is the statistical evaluation of the survey results to verify the data are the correct type, quality, and quantity to support the conclusion (Reference 1, Section 8.1). The DQA is a continuation of the DQO process, and verifies that the objectives of the DQOs have been met. As with the DQOs, the DQA is tailored to the desired goals of the specific site or project. For sites with very low residual radioactivity, a formal statistical review may not be required, given that all results of the survey are less than the $DCGL_w$. For other sites, the DQA can provide valuable insights as to the appropriateness of the selected survey design. Accordingly, the DQA should be performed as soon as practicable following the completion of the survey, especially for large sites where data may be collected over a period of months or years.

The evaluation of survey results is performed as part of the DQA process. The final step of the DQA includes a comparison of survey results against the defined release criteria.

5.4.1 Review the Data Quality Objectives and Survey Design

The initial step of the DQA process is to evaluate the data against the criteria outlined in the survey design and against the established DQOs. The comparison against the established DQOs typically includes verifying that the appropriate number of measurements have been collected and ensuring that the desired power for the statistical test is achieved. The power of the test is defined as the probability of committing a Type II decision error (i.e., accepting the null hypothesis when it is false) and is calculated as $1-\beta$ (refer to Section 5.2.2).

A useful tool in the review of these two factors is a post-survey/sample calculation of the required number of measurements, using the actual survey unit standard deviation. A simpler review exercise would be to ensure that the actual standard deviation is less than the assumed standard deviation, thus confirming that the actual required minimum number of measurements was collected to support the initial survey design. This indicator is much less complicated than generating a power curve and achieves the same result given that the accuracy of power depends on the estimate of σ and the number of measurements. However, a continued failure of the statistical test indicates the need to evaluate survey design. The power curve is a useful tool for this purpose.

A more detailed indicator of power may be desired, especially if the levels of residual radioactivity at the site are near the $DCGL_w$. For this case, the development of a power curve can be useful. While the power curve can be generated prospectively (prior to the collection of data), the user should recognize that the accuracy of the curve will depend on the estimates of data variability, the number of measurements, and the standard deviation (σ). Thus, the user may find a prospective power curve to be more useful. Instructions on developing a power curve are provided in Appendix I of MARSSIM. A more recent reference that provides a simplified discussion of the power curve is presented by Duvall in Reference 57.

5.4.2 Conduct a Preliminary Data Review

The preliminary data review involves the review of quality assurance reports and statistical and graphical evaluation of the data. The calculation of basic statistical quantities, such as the mean, standard deviation, and median, is useful for comparison against the assumptions used in the survey design and acceptance criteria. For example, the calculated standard deviation should be less than the value assumed in calculating the required number of measurements. In addition, the comparison of the mean of the data against the reference area average and $DCGL_W$ provides a preliminary indication of the outcome of the statistical test. The median serves as an indicator of the skewness of the data (Reference 1, Section 8.2.2). If a large variation between the mean and median exists, the data are skewed.

Other parameters include the maximum, minimum, and range. The maximum result must be less than the $DGGL_{EMC}$.

A graphical data review should almost always include a posting plot and a histogram (Reference 1, Section 8.2.2). The posting plot is a map of the survey unit with the data entered at the respective locations, and provides information regarding trends in the data (i.e., the area extent of residual activity in the survey unit). The histogram (or frequency plot) describes the shape of the data distribution and provides information regarding the skewness or symmetry of the data.

The histogram, in some cases, may provide the appropriate background for a given survey unit. This is especially valuable for sites where backgrounds vary greatly from location to location. Additional discussion regarding alternate methods for background determination is provided in Section 5.1.1.

Other graphical representations include scatter plots, confidence intervals, ranked data plots, quantile plots, stem-and-leaf diagrams, and spatial or temporal plots. A detailed discussion of each graph type is provided in Appendix I of MARSSIM.

5.4.3 Select the Statistical Tests

As discussed in Section 5.2.1, the decision to use a particular statistical test is typically made during the planning phases. The preliminary data review provides assurance that the survey design is appropriate, based on an evaluation of the assumptions for the test. The nonparametric tests recommended by MARSSIM, and discussed in Section 5.2.1 of this document, are tests of the median concentration of residual radioactivity in the survey unit. However, the use of the nonparametric test does not assume a given distribution for the data set. Thus, if the data are symmetrical, then the test of the median is also a test of the mean. If the data are not symmetrical, then the appropriate decision will still be made about whether or not the mean concentration exceeds the $DCGL_W$. The nonparametric tests are considered “robust”, as they are applicable over a wide range of conditions. The user should also recognize the fact that a parametric test designed for an exact set of conditions will have higher power. The objective is to select a statistical test that fits the assumptions for a given site.

5.4.4 Verify Assumptions of the Test

The primary advantage in the selection of the nonparametric tests is that their assumptions are minimal. The two assumptions that must be verified include (Reference 1, Section 8.2.4):

- The data from the reference area and the survey unit must consist of independent samples from each distribution, and

- An adequate number of measurements/samples is collected

If a parametric test is selected, the applicable assumptions must be verified. The following table provides some examples of assumptions and tools for verifying the assumptions:

Table 5-4
Methods for Checking the Assumptions of Statistical Tests[1]

Assumption	Diagnostic Tool
Spatial Independence	Posting Plot
Symmetry	Histogram, Quantile Plot
Data Variance	Sample Standard Deviation
Power is Adequate	Retrospective Power Chart

[1] MARSSIM Table 8.1

5.4.5 Draw Conclusions from the Data

The final step of the DQA process is to draw a conclusion as to whether or not the survey unit meets the defined release criteria. Note that the comparison of data to the Investigation Levels (refer to Table 5-3) has already taken place during a preliminary review of the data. Assuming that no investigation criteria have been exceeded, the final evaluation is performed to determine if the survey unit meets the release criteria.

Table 5-5 provides an example of the conclusions that can be drawn for given conditions.

Table 5-5
Evaluating Survey Results

Comparison	Disposition
Compare individual measurements and sample concentrations to the $DCGL_{W.}$	If all results are less than the $DCGL_W$, then the survey units meets the release criteria.
Compare individual measurements and sample concentrations to the $DCGL_{EMC}$ for evidence of small areas of elevated activity	If any value exceeds $DCGL_{EMC}$, the survey unit does not meet the release criteria.
Perform statistical test on data to verify survey unit average is less than $DCGL_{W.}$	If statistical test fails, the survey unit does not meet the release criteria.
Perform hot spot unity rule test (Reference 1, Equation 8-2).	If unity rule fails, the survey unit does not meet the release criteria.

In summary, there are three conditions that must be satisfied for the survey unit to be acceptable for release:

1) Statistical test passes: survey unit average is less than the $DCGL_W$.

2) Every hot spot is less than the $DCGL_{EMC}$.

3) The hot spot unity rule has been satisfied (Reference 1, Equation 8-2).

5.5 Data Presentation

Documentation for the Final Status Survey typically includes data packages for individual survey units, which include the survey instructions for the survey unit, the resulting data, and a final report that includes a summary of the data and the conclusions drawn from the data. It is recommended that the site retains the individual data packages. The final report, which should include the appropriate level of detail to allow for a verification of the data, is submitted to the appropriate regulatory agency for review and approval.

The format for the final report should be described in the Final Status Survey plan in order to avoid delays in report review and approval.

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A
EXAMPLE OF A SAMPLING PLAN FOR SUBFOUNDATION SOIL

Background

The design of sampling plans is a highly project-specific task. Any surface soil sampling plan should address the guidance provided by MARSSIM. This plan was provided for soil beneath a building with a prior history of containing a radioactive spill. References to site-specific procedures and standard project methods are typically included in such plans. Sampling plans should include detailed sample locations and may also include a brief description of the history that was used to design the sampling approach. Field conditions may dictate that plans be amended, as needed in the field to accomplish objectives.

The example provided here is for sampling of subsurface soil, a medium not addressed by MARSSIM.

SUBFLOOR CHARACTERIZATION SAMPLING PLAN
Safe Shutdown System (SSS) Building Areas YS002, YS003, YS004

References

1. Project Procedure for Collection of Subsurface Soil Sampling for Site Characterization
2. Project Procedure for Collection of Site Characterization and YS Samples
3. Project Procedure for Sample Custody and Control
4. Project Procedure for Preparation of Job Hazards Checklist
5. Project Technical Basis Document to Determine Radiological Background Levels for Soil

Description

The SSS Building is located in the SW corner of the plant yard area. The building was constructed in 1982. The components in the building were designed to provide a makeup for primary and secondary plant systems after a flood, seismic event, tornado or similar occurrence. The SSS system was also designed to bring the plant to a cold shutdown during a transient situation such as a fire in the Control Room or Turbine Building.

The SSS Building is comprised of three rooms separated by concrete block walls. Each room is considered a separate Final Status Survey area and is designated accordingly for subfoundation soil characterization. All electrical power to this building has been terminated.

The SSS Building south room (YS002) contained the SSS diesel generator and battery rack and two above-ground diesel fuel tanks. The diesel generator and battery racks have been removed and the two fuel tanks remain. The fuel tanks are housed within a single concrete secondary container located in the northwest corner of this room. The room also contains multiple embedded conduits that run beneath the concrete floor.

The SSS Building Center Room (YS003) contained the electrical power supply along the south wall. The area also contains embedded conduit that runs beneath the concrete floor from the SSS Motor Control Center that previously resided along the south wall and the instrument panel near the center of the room.

The SSS Building North Room (YS004) contained the primary pump, the secondary pump and the Boron Addition Tank. The north end of the room contains a vertical pipe chase/sump through the room floor; pipes for the various systems (i.e. SI Tank, Fire Protection Water Tank) enter and exit through this chase to the subsurface. A floor sump is approximately eight feet deep and it currently contains a few inches of standing water.

History

In 1982 approximately 50 gallons (0.2 m^3) of water leaked from the plant's Purification system into the SI system and into the SSS building (Yard Survey Area YS004) adjacent the south side of the pipe chase. The contaminated water spread over the floor to the YS003 area through a doorway and flowed across an expansion joint adjoining the concrete floors of the two areas. This water contained approximately 150 µCi (5.55 E6 Bq) of mixed fission products. Survey information from the time of the spill indicated that the contamination was contained within the SSS building and limited to the floor and lower walls. Removable contamination survey results were as high as 52,000 dpm/100cm^2 (868 Bq/100 cm^2) and direct frisk results were as high as 15,000 cpm. The water was transferred to the Waste Disposal Building for processing and the room was decontaminated to less than 1000 dpm/100cm^2 (16.7 Bq/100 cm^2). The concrete floor was coated with a sealant after decontamination.

As part of the subsurface pipe assessment program the pipe chase was surveyed with swipes producing up to 2,000 gross dpm (33.4 Bq). The concrete surface of the pipe chase was decontaminated and about a foot of soil was remediated beneath the exit point from the building.

Drawings

Attached Floor Plans YS002, YS003, YS004 (Figure A-1)

Plant Drawings of SSS Building

Plant Subsurface Utilities Drawing

Sampling Objective and Approach

The objective of this sampling plan is to obtain targeted and randomly located soil samples from beneath the foundation of the SSS building. An initial target sampling strategy for this subfoundation soil location is based on the project Subfoundation Sampling Decision Matrix (Figure A-2). This matrix was devised to address plant areas where there had been buildings constructed since the start of plant operation with some potential for radioactivity below their foundations. General sampling for this purpose is based on NUREG/CR-5849.

Sub-foundation sampling of soils can be warranted for a site based on its history (see Main Report, Section 2.4.4.3.4). In this case information indicates that for certain locations sub-foundation soils may contain radioactivity due to:

- Leakage of a source term through or near a foundation or
- Placement of a new foundation on ground previously exposed to plant activity.

For original plant structures (built at the time of plant construction) only leakage near or through the foundation should be an issue. Erection of a new structure's foundation may have occurred on soil whose radiological status is now in question due to historical changes in detection capabilities or assessment criteria that has been modified since construction.

Figure A-2 is a flow chart used to provide a general determination of sub-foundation sampling needs. It shows that structures are categorized based on the time of their construction. General sampling is then planned and performed accordingly using the site history as guidance. Finally, if radioactivity is detected, additional sampling will be planned based on the initial results.

Soil samples will be collected through holes drilled in the concrete floors of the building. Soil samples will be collected to a depth necessary to identify the limits of any contamination. See attached drawings for sample locations.

Gamma spectroscopy and Total Petroleum Hydrocarbon (TPH) analyses are planned for the soil samples. Consult sampling procedures for sample collection methods and shipping timing and container requirements.

Radiological and Safety Preparations

No Radiological Work Permit (RWP) is required to enter the subject building. Samples are not expected to contain significant levels of radioactivity. However, protective measures indicated in sampling procedures should be followed explicitly, including the use of protective clothing, as needed. All subfloor soil sampling should be preceded by a thorough frisk of the hole using hand-held instruments. Per project procedures, any excessive activity levels should be reported to radiological protection personnel prior to commencing sampling.

Notifications

Prior to the start of work notify the following:

- Engineering Department
- Control Room
- Safety Department
- Radiation Protection

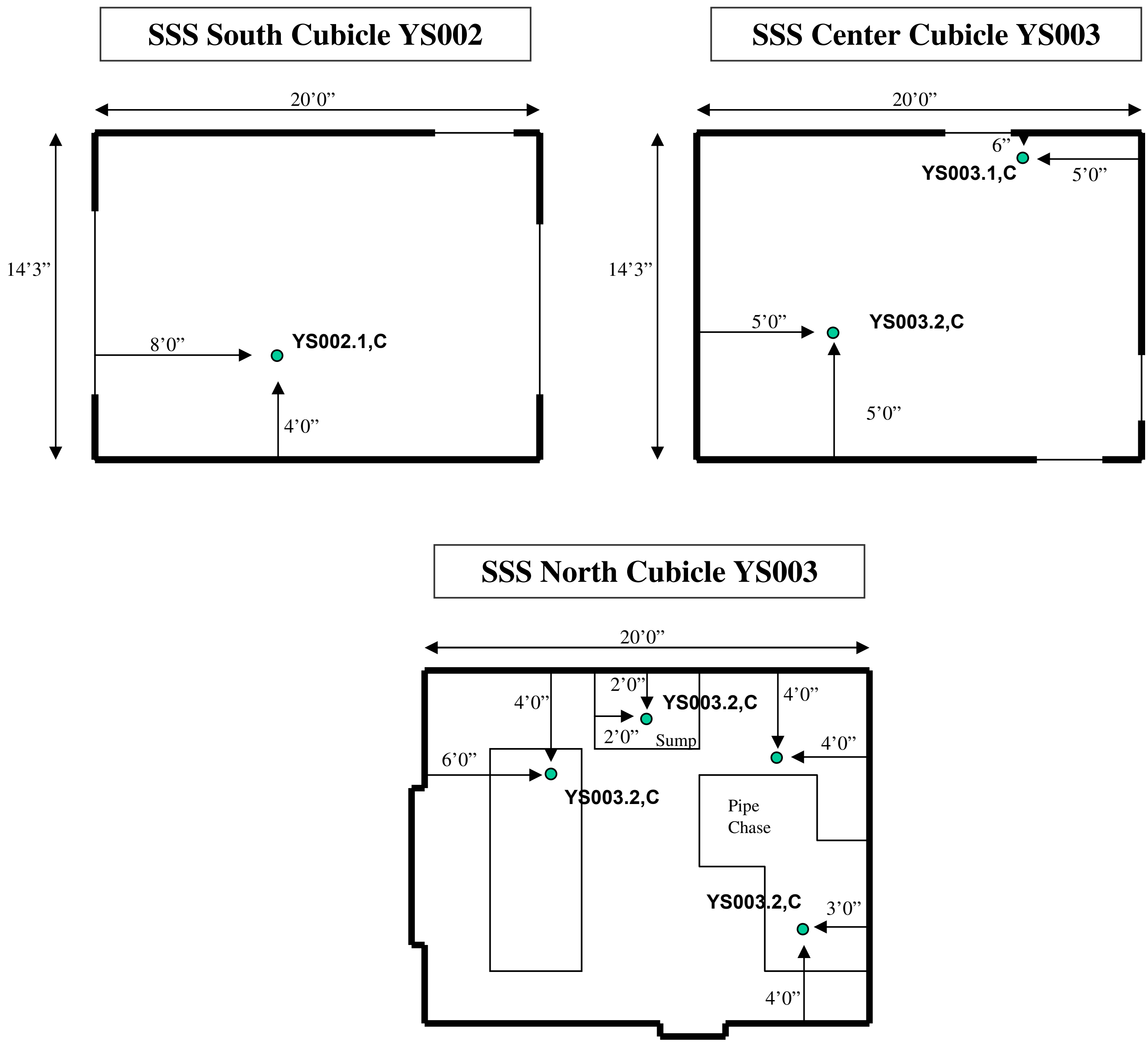

Figure A-1
SSS Building Floor Plans and Sample Locations

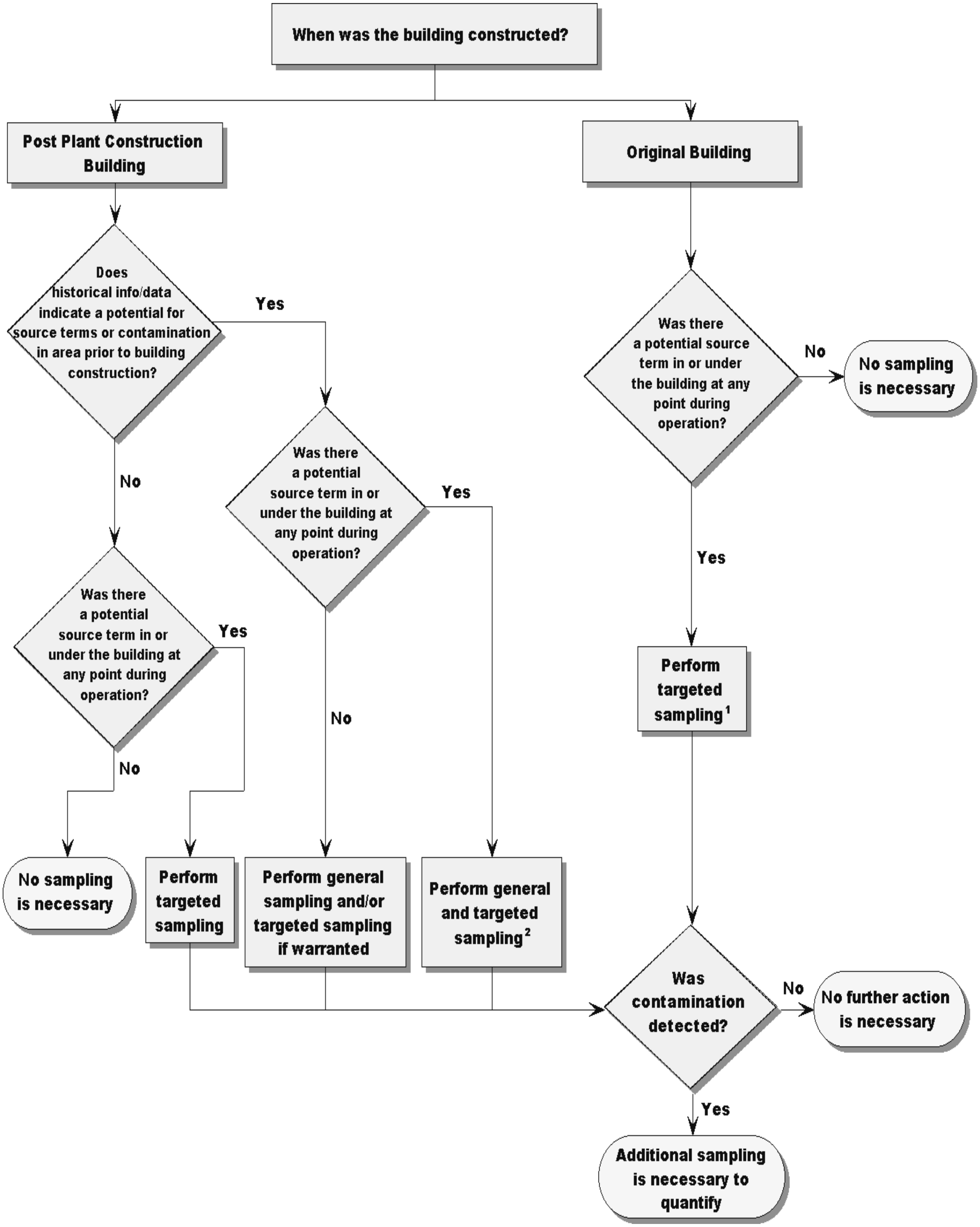

Figure A-2
Subfoundation Sampling Decision

B
CONCEPTUAL SCHEDULE FOR SITE CHARACTERIZATION

Conceptual Schedule for Site Characterization

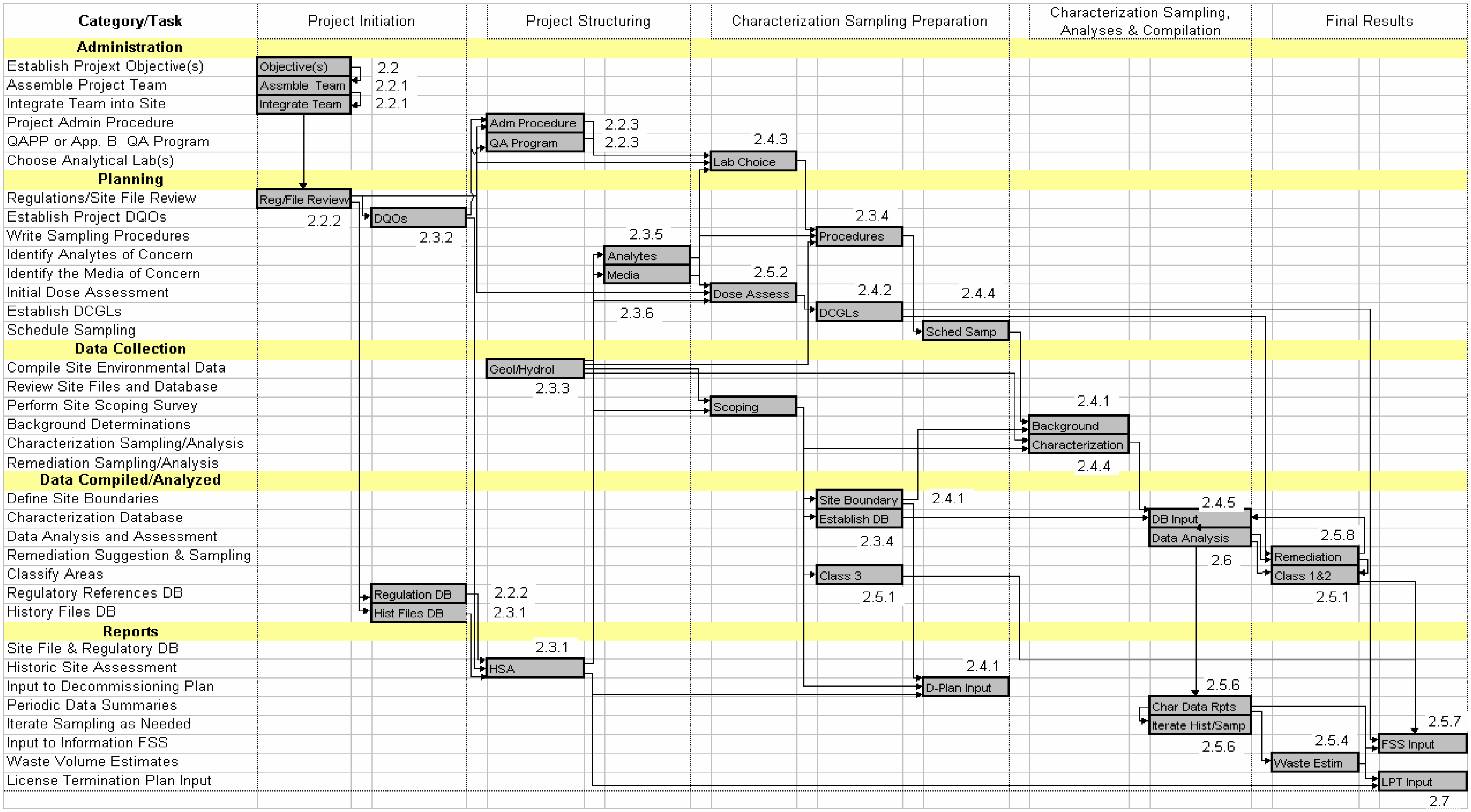

Figure B-1
Conceptual Schedule for Site Characterization (cross-referenced to report sections)

Target:
Nuclear Power

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