U.S.NRC
United States Nuclear Regulatory Commission
*Protecting People and the Environment*

NUREG-1757
Volume 2
Revision 2

# Consolidated Decommissioning Guidance

## Characterization, Survey, and Determination of Radiological Criteria

## Draft Report for Comment

**Office of Nuclear Material Safety and Safeguards**

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# Consolidated Decommissioning Guidance

## Characterization, Survey, and Determination of Radiological Criteria

## Draft Report for Comment

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Prepared by:
C.S. Barr, S. Clark,
G.C. Chapman, J.P. Clements,
D.W. Esh, R.W. Fedors,
A.M. Huffert, L.A. Kauffman,
M.M. LaFranzo, D.B. Lowman,
C.A. McKenney, L.L. Parks,
D.W. Schmidt*, A.L. Schwartzman,
and B.A. Watson
*retired

Cynthia Barr, NRC Project Manager

Office of Nuclear Material Safety and Safeguards

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Any interested party may submit comments on this report for consideration by the NRC staff. Comments may be accompanied by additional relevant information or supporting data. Please specify the report number NUREG-1757, Volume 2, Revision 2 in your comments, and send them by the end of the comment period specified in the *Federal Register* notice announcing the availability of this report.

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**Federal Rulemaking Website:** Go to http://www.regulations.gov and search for documents filed under Docket ID **NRC-2020-0192**.

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For any questions about the material in this report, please contact: Cynthia Barr, Senior Risk Analyst, 301-415-4015 or by e-mail at cynthia.barr@nrc.gov.

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This NUREG provides voluntary guidance for implementing the mandatory information collections in 10 CFR 20 that are subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et. seq.). These information collections were approved by the Office of Management and Budget (OMB), approval number 3150-0014. Send comments regarding this information collection to the FOIA, Library, and Information Collections Branch (T-6A10), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by e-mail to Infocollects.Resource@nrc.gov, and to the OMB reviewer at: OMB Office of Information and Regulatory Affairs (3150-0014), Attn: Desk Officer for the Nuclear Regulatory Commission, 725 17th Street, NW Washington, DC 20503; e-mail: oira_submission@omb.eop.gov.

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# ABSTRACT

The U.S. Nuclear Regulatory Commission (NRC), Office of Nuclear Material Safety and Safeguards (NMSS), previously consolidated and updated numerous decommissioning guidance documents into a three-volume NUREG. This NUREG series is intended for use by the NRC staff, licensees, and others. The three volumes address the following topics:

(1) Decommissioning Process for Materials Licensees

(2) Characterization, Survey, and Determination of Radiological Criteria

(3) Financial Assurance, Recordkeeping, and Timeliness

The staff last updated Volume 2 of the NUREG series, entitled, "Consolidated Decommissioning Guidance: Characterization, Survey, and Determination of Radiological Criteria," in September 2006. This volume provides guidance on compliance with the radiological criteria for license termination (License Termination Rule (LTR)) in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 20, "Standards for Protection against Radiation," Subpart E, "Radiological Criteria for License Termination." This guidance takes a risk-informed, performance-based approach to the demonstration of compliance. This guidance will help identify the information (subject matter and level of detail) needed to terminate a license and considers the specific circumstances of the wide range of NRC licensees. Licensees should use this guidance in preparing decommissioning plans (DPs), license termination plans, final status surveys, and other technical decommissioning reports for NRC submittal. The NRC staff will use the guidance in reviewing these documents and related license amendment requests. Volume 2 applies to all licensees subject to the LTR (i.e., fuel cycle, fuel storage, materials, and reactor licensees).

Changes made to this revision of Volume 2 include the following:

- **Dose Modeling**—adds guidance on model abstraction and simplification, consideration of elevated areas, use of distribution coefficients in dose modeling, and consideration of uncertainty.

- **"As Low As Is Reasonably Achievable (ALARA)" Analysis**—updates guidance on the ALARA analysis review particularly for restricted release based on lessons learned from proposed restricted release scenarios.

- **Composite Sampling**—adds information on methodologies for incorporating composite sampling strategies into final status survey plans and details when it would or would not be appropriate to use composite sampling.

- **Characterization**—updates information in Appendix F on surface water and groundwater characterization.

- **Engineered Barrier Analysis**—updates and reorganizes Section 3.5 and Appendix P; includes new information on how ALARA is considered prior to engineered barriers for restricted release and updates bibliography with new references on evaluation of engineered performance and degradation.

- **Radiological Surveys**—provides updated guidance on subsurface radiological surveys, including surveys associated with excavations and re-use of soils; and provides additional information on use and implementation of Scenario B.

- **Lessons Learned**—removes Appendix O which contains lessons learned from RIS-2002-02; and archives questions and answers, and lessons learned in ADAMS (ADAMS Accession Number (No.) ML20052C815). Future interim guidance will be placed on the NRC public decommissioning website to allow for more timely updates to guidance between NUREG revisions. Frequently asked questions can also be found in NUREG-1628.

- **Miscellaneous Editorial Changes**–corrects typographical and formatting errors; adds clarity to areas of the guidance.

# FOREWORD

**The NRC staff suggests that licensees contact the NRC or the appropriate Agreement State authority to ensure understanding of what actions should be taken to initiate and complete decommissioning at facilities.**

In September 2003, the U.S. Nuclear Regulatory Commission (NRC) staff in the Office of Nuclear Material Safety and Safeguards (NMSS) consolidated and updated the policies and guidance of its decommissioning program in a three-volume NUREG series, NUREG-1757, "Consolidated NMSS Decommissioning Guidance." This NUREG series provides guidance on planning and implementing license termination under the NRC's License Termination Rule (LTR) in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 20, "Standards for Protection against Radiation," Subpart E, "Radiological Criteria for License Termination"; complying with the radiological criteria for license termination; and complying with the requirements for financial assurance and recordkeeping for decommissioning and timeliness in decommissioning materials facilities. The staff periodically updates NUREG-1757 to reflect current NRC decommissioning policy.

In September 2005, the staff issued, for public comment, draft Supplement 1 to NUREG-1757, which contained proposed updates to the three volumes of NUREG-1757. Draft Supplement 1 included new and revised decommissioning guidance that addresses some of the LTR implementation issues, which were analyzed by the staff in two Commission papers (SECY-03-0069, "Results of the LTR Analysis," dated May 2, 2003; and SECY-04-0035, "Results of the License Termination Rule Analysis of the Use of Intentional Mixing of Contaminated Soil," dated March 1, 2004). These issues include restricted use and institutional controls, onsite disposal of radioactive materials (10 CFR 20.2002, "Method for Obtaining Approval of Proposed Disposal Procedures"), selection and justification of exposure scenarios based on reasonably foreseeable future land use (realistic exposure scenarios), intentional mixing of contaminated soil, and removal of material after license termination. The staff also developed new and revised guidance on other issues, including engineered barriers.

The staff received stakeholder comments on the draft NUREG and prepared responses to these comments. Stakeholder comments and responses are located in the Agencywide Documents Access and Management System at Accession Number ML062370521. Comments were addressed and updated sections from Supplement 1 were placed into the appropriate locations in revisions of Volumes 1 and 2 of NUREG-1757 (NUREG-1757, Volume 1, Revision 2; and NUREG-1757, Volume 2, Revision 1). The staff plans to revise Volume 3 of this NUREG series at a later date, and that revision will incorporate the Supplement 1 guidance that is related to Volume 3.

The NRC has increased the use of risk information in its regulation of nuclear materials and nuclear waste management, including the decommissioning of nuclear facilities. The NRC's risk-informed regulatory approach to the decommissioning of nuclear facilities represents a philosophy whereby risk insights are considered, together with other factors, to better focus the attention and resources of both the licensee and the NRC on the more risk-significant aspects of the decommissioning process and on the elements of the facility and the site that will most

affect risk to members of the public following decommissioning. This results in a more effective and efficient regulatory process.

The information used to "risk inform" the decommissioning process typically comes from the results and findings of risk assessments or dose modeling. A risk assessment is a type of systematic analysis used to understand what can happen, how likely it is to happen, and the resulting consequences. Dose modeling is used to estimate potential dose to members of the public who may use the decommissioned site in the future following license termination. The end result of such assessments (e.g., the calculation of predicted doses from decommissioned sites) relates directly or indirectly to public health effects. The NRC staff has developed this guidance consistent with a risk-informed approach.

The primary decommissioning guidance documents used by licensees and the NRC staff are NUREG-1757 and NUREG-1700, Revision 2, "Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans," issued April 2018. NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors," issued February 1996, includes a section on decommissioning and license termination for nonpower reactors. Table 1 below describes the general applicability of these documents.

Since the last revision of this volume in 2006, which addressed comments on draft Supplement 1, the NRC staff and larger decommissioning community has gained experience on a number of technical issues for which guidance has been developed. Table 2 describes the most significant changes to the guidance in this volume to include new and updated information based on this experience.

NUREG-1757, Volume 2, Revision 2, is applicable to all licensees that are subject to the LTR. NUREG-1757 is intended for use by applicants, licensees, NRC license reviewers, and other NRC personnel. It is also available to Agreement States and the public.

This NUREG is not a substitute for NRC regulations, and compliance with it is not required. The NUREG describes approaches that are acceptable to the NRC staff. However, methods and solutions different than those in this NUREG will be acceptable, if they provide a basis for concluding that the decommissioning actions are in compliance with NRC regulations.

_____
Patricia Holahan, Director
Division of Decommissioning, Uranium Recovery, and Waste Programs
Office of Nuclear Material Safety and Safeguards

**Table 1 Contents and Applicability of Key Decommissioning Guidance Documents**

| Volume and Status[1] | Title | Licensees to Which the Guidance Applies |
|---|---|---|
| NUREG-1757, Vol. 1, Rev. 2; September 2006 | Consolidated Decommissioning Guidance: Decommissioning Process for Materials Licensees | Fuel cycle, fuel storage, and materials licensees[2]; limited applicability to reactor licensees (see text below) |
| NUREG-1757, Vol. 2, Rev. 2; November 2020 (this revision) | Consolidated Decommissioning Guidance: Characterization, Survey, and Determination of Radiological Criteria | All licensees that are subject to the LTR (fuel cycle, fuel storage, materials, and reactor licensees) |
| Draft NUREG-1757, Vol. 3, Rev. 1, February 2012 | Consolidated Decommissioning Guidance: Financial Assurance, Recordkeeping, and Timeliness | Volume 3 is intended to apply only to the decommissioning of materials facilities licensed under Title 10 of the Code of Federal Regulations (10 CFR) Parts 30, 40, 70, and 72. |
| NUREG-1700, Rev. 2; April 2018 | Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans | Power reactor licensees |
| NUREG-1537, February 1996 | Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors | Nonpower reactor licensees |

Notes:

[1] Versions listed are current as of the date of publication of this document in 2020. The NRC's Library at http://www.nrc.gov/reading-rm/doc-collections/nuregs contains the most up-to-date version.

[2] This refers to licensees regulated under 10 CFR Parts 30, 40, 60, 61, 63, 70, and 72 (for 10 CFR Parts 60, 61, and 63, only the ancillary surface facilities that support radioactive waste disposal activities). Because uranium recovery facilities are not subject to 10 CFR Part 20, Subpart E, NUREG-1620, Revision 1, Section 5, should be used for decommissioning guidance for uranium recovery facilities that are subject to Appendix A, "Criteria Relating to the Operation of Uranium Mills and the Disposition of Tailings or Wastes Produced by the Extraction or Concentration of Source Material From Ores Processed Primarily for Their Source Material Content," to 10 CFR Part 40, "Domestic Licensing of Source Material."

**Table 2 Summary of Major Changes to Volume 2, Revision 2**

| Subject | Affected Sections |
|---|---|
| Updated guidance on review of dose modeling used to demonstrate compliance with radiological criteria for license termination including the following topics:<br>• revised and added guidance on technical issues associated with model abstraction and simplification,<br>• source term abstraction and development of site-specific parameters such as distribution coefficients, and<br>• consideration of elevated areas or "hot spots". | Chapters 2, 5, Appendix I |
| Updated guidance on engineered barriers to reflect recent research; updated guidance on consideration of ALARA prior to use of engineered barriers to reduce dose for restricted release. | Section 3.5 and Appendix P |
| Updated guidance on review of ALARA analysis including experience and lessons learned gained from proposed restricted release scenarios; updated regulatory citations and guidance related to discounting and the monetary value of collective dose averted. | Chapter 6 and Appendix N |
| Updated guidance on surface water and groundwater characterization. | Appendix F |
| Added guidance on subsurface investigations, survey of excavations, survey of back-fill soils, data visualization tools, integration of dose modeling and radiological surveys, and implementation of Scenario B. | Section 3.6 and Appendix G |
| Updated guidance and information related to use of screening values and resuspension factors. Cites NUREG/CR-5512, Volume 3 for screening values for additional radionuclides. | Appendix H |
| Streamlined guidance on consideration of buried radioactivity. Added guidance on intrusion events and exposure scenarios for large substructures. | Appendix J |
| Streamlined guidance on site-specific exposure scenarios. Updated guidance on regulatory standards. | Appendix M |
| Added guidance on use of composite sampling including information on when it would and would not be appropriate to use, derivation of modified investigation levels, and methods to incorporate composite sampling into survey designs. | Appendix O |
| Added guidance on consideration of uncertainty in performance assessment analyses including issues associated with use of generic data sets, unrepresentative data, model integration, risk dilution, and lack of correlation of correlated parameters. | Appendix Q |

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# ACKNOWLEDGMENTS

The preparers would like to thank the following individuals for assisting in the development and review of this revision to the guidance document. These individuals provided valuable contributions, insights, and recommendations. A special thanks to Lifeng Guo for his contributions to Appendix F, “Surface Water and Groundwater Characterization”; Mark Fuhrmann and Hans Arlt for their contributions to various sections of the document including guidance on the use of engineered barriers in Section 3.5 and Appendix P; and Allen Gross for his contributions to Appendix M, “Process for Developing Exposure Scenarios using Site Specific Information”. The preparers would also like to acknowledge the contribution of Michael Kunowski for sharing his valuable NRC regional inspector experience, which greatly enhanced the quality of the guidance document. The preparers would like to acknowledge the very thorough review performed by Sheldon Clark in the NRC’s Office of General Counsel whose edits and comments substantially improved the defensibility and readability of the document. Comments received from the States of California, Pennsylvania, and Texas were also invaluable to obtain Agreement State perspectives and greatly improved the clarity and quality of the document. Finally, a special thanks goes out to Marcia Pringle and Sarah Achten for technical editing and formatting of the document without which this document could not be published.

## Revision 2 Experts and Peer Reviewers

H. Arlt, M Fuhrmann, A. Gross,
L. Guo, and M. Kunowski

## Revision 1 Preparers (not including preparers of Revision 2)

### U.S. Nuclear Regulatory Commission Staff

K.L. Banovac, J.T. Buckley,
R.L. Johnson, J.J. Kottan,
T.G. McLaughlin, and S. Schneider

### Agreement States

California, Pennsylvania, and Texas

# ABBREVIATIONS

| | |
|---|---|
| ACAP | Alternative Cover Assessment Program |
| ADAMS | Agencywide Documents Access and Management System |
| AEA | Atomic Energy Act of 1954, as amended |
| ALARA | as low as is reasonably achievable |
| ALCD | Alternative Landfill Cover Demonstration |
| Am | americium |
| ANSI | American National Standards Institute |
| ASME | American Society of Mechanical Engineers |
| ASR | alkali-silica reaction |
| ASTM | American Society for Testing and Materials |
| Bq | becquerel |
| Bq/kg | becquerel per kilogram |
| Ca | calcium |
| CERCLA | Comprehensive Environmental Response, Compensation, and Liability Act |
| CFR | *Code of Federal Regulations* |
| Ci | curie |
| CNWRA | Center for Nuclear Waste Regulatory Analyses |
| Co | cobalt |
| $CO_2$ | carbon dioxide |
| cpm | counts per minute |
| Cs | cesium |
| C-S-H | calcium silicate hydrate |
| CSM | Conceptual site model |
| DandD | Decontamination and Decommissioning software package |
| DCGL | derived concentration guideline level |
| $DCGL_{EMC}$ | derived concentration guideline level (elevated measurement comparison) |
| $DCGL_W$ | derived concentration guideline level (wide-area [over the entire survey unit]) |
| DP | decommissioning plan |
| dpm | disintegrations per minute |
| DQA | data quality assessment |
| DQOs | data quality objectives |
| DQO process | data quality objectives process |
| Eh | redox potential |
| EMC | elevated measurement comparison |
| EML | DOE Environmental Measurements Laboratory (formerly the Health and Safety Laboratory) |
| EPA | U.S. Environmental Protection Agency |

| | | |
|---|---|---|
| Fe | iron |
| FEP | feature, event, and process |
| FR | *Federal Register* |
| FSS | Final status survey |
| FSSP | Final status survey plan |
| FSSR | Final status survey report |
| GCL | geosynthetic clay liner |
| GIS | geographic information system |
| H | hydrogen |
| H-3 | tritium |
| HDPE | high-density polyethylene |
| HSA | historical site assessment |
| HTD | hard-to-detect |
| IAEA | International Atomic Energy Agency |
| ICRP | International Commission on Radiological Protection |
| IL | investigation level |
| ITRC | Interstate Technology and Regulatory Council |
| ISO | International Organization for Standardization |
| $K_d$ | distribution coefficient |
| kg | kilogram |
| km | kilometer |
| L | liter |
| L/kg | liter/kilgoram |
| LBGR | lower bound of the gray region |
| LLW | low-level waste |
| LTP | License termination plan |
| LTR | License termination rule |
| MARLAP | Multi-Agency Radiological Laboratory Analytical Protocols Manual (NUREG-1576) |
| MARSSIM | Multi-Agency Radiological Survey and Site Investigation Manual (NUREG-1575) |
| MCL | maximum contaminant level |
| MDC | minimum detectable concentration |
| $MDC_{scan}$ | scan minimum detectable concentration |
| $MDC_{static}$ | static minimum detectable concentration |
| Mg | magnesium |
| mg | milligram |
| mil | unit of corrosion (1 mil=0.001 inch) |
| MIL | modified investigation level |
| mrem | millirem |
| mSv | millisievert |

| | |
|---|---|
| Na | sodium |
| NAS | National Academy of Sciences |
| NCRP | National Council on Radiation Protection and Measurements |
| NEPA | National Environmental Policy Act |
| Ni | nickel |
| NIST | National Institute of Standards and Technology |
| NMSS | Office of Nuclear Material Safety and Safeguards (U.S. Nuclear Regulatory Commission) |
| NOAA | National Oceanic and Atmospheric Administration |
| NORM | naturally occurring radioactive material |
| Np | neptunium |
| NRC | U.S. Nuclear Regulatory Commission |
| NUREG | NRC technical report designation |
| Pb | lead |
| pCi | picocurie |
| pCi/g | picocuries per gram |
| PDF | probability density function |
| pH | hydrogen ion concentration (negative of the log of the hydrogen ion molar concentration) |
| Po | polonium |
| PSR | partial site release |
| Pu | plutonium |
| PVC | polyvinyl chloride |
| QA | quality assurance |
| QAPP | quality assurance project plan |
| QA/QC | quality assurance and quality control |
| Ra | radium |
| RCRA | Resource Conservation and Recovery Act |
| REMP | Radiological Environmental Monitoring Program |
| RESRAD | RESidual RADdioactive materials computer code |
| RFR (also Rfo) | resuspension factor |
| ROC | radionuclide(s) of concern |
| RSS | residual sum of squares |
| RSSI | radiation survey and site investigation (process) |
| SADA | Spatial Analysis and Decision Assistance (computer code) |
| SDMP | Site Decommissioning Management Plan |
| Sr | strontium |
| SRM | staff requirements memorandum |
| SRP | NMSS Decommissioning Standard Review Plan (NUREG-1727) |
| Sv | sievert |
| TEDE | total effective dose equivalent |

| | | |
|---|---|---|
| Th | thorium |
| U | uranium |
| $UF_6$ | uranium hexafluoride |
| UMTRA | Uranium Mill Tailings Remedial Action |
| UMTRCA | Uranium Mill Tailings Radiation Control Act |
| $UO_2$ | uranium dioxide |
| USDA | U.S. Department of Agriculture |
| U.S.C. | U.S. Code |
| USGS | U.S. Geological Survey |
| WRS | Wilcoxon Rank Sum (test) |
| xLPR | Extremely Low Probability of Rupture (project) |
| y | year |

# GLOSSARY

The following terms are defined for the purposes of this volume of the NUREG report.

*Affected Parties*. Representatives of a broad cross section of individuals and institutions in the community or vicinity of a site that may be affected by the decommissioning of the site.

*Acceptance Review*. The evaluation the NRC staff performs upon receipt of a license amendment request to determine if the information provided in the document is sufficient to begin the technical review.

*Activity*. The rate of disintegration (transformation) or decay of radioactive material. The units of activity are the curie (Ci) and the becquerel (Bq) (see Title 10 of *the Code of Federal Regulations* (10 CFR) 20.1003, "Definitions").

*ALARA*. The acronym for "as low as is reasonably achievable," which means making every reasonable effort to maintain exposures to radiation as far below the dose limits as is practical, consistent with the purpose for which the licensed activity is undertaken, and taking into account the state of technology, the economics of improvements in relation to the state of technology, the economics of improvements in relation to the benefits to public health and safety, and other societal and socioeconomic considerations, and in relation to the use of nuclear energy and licensed materials in the public interest (see 10 CFR 20.1003).

*Alternate Criteria*. Dose criteria for residual radioactivity that are greater than certain dose criteria described in 10 CFR 20.1402, "Radiological Criteria for Restricted Use," and 10 CFR 20.1403, "Criteria for License Termination under Restricted Conditions," as allowed in 10 CFR 20.1404, "Alternate Criteria for License Termination." Alternate criteria must be approved by the Commission.

*Aquifer*. A geologic formation, a group of formations, or part of a formation capable of yielding a significant amount of groundwater to wells or springs.

*Background Radiation*. Radiation from cosmic sources, naturally occurring radioactive material, including radon (except as a decay product of source or special nuclear material) and global fallout as it exists in the environment from the testing of nuclear explosive devices or from past nuclear accidents, such as Chernobyl, that contribute to background radiation and are not under the control of the licensee. Background radiation does not include radiation from source, byproduct, or special nuclear materials regulated by the NRC (see 10 CFR 20.1003).

*Broad-Scope Licenses*. A type of specific license authorizing receipt, acquisition, ownership, possession, use, and transfer of any chemical or physical form of the byproduct material specified in the license but not exceeding quantities specified in the license. Relevant requirements are found in 10 CFR Part 33, "Specific Domestic Licenses of Broad Scope for Byproduct Material." Examples of broad scope licensees are large universities and large research and development facilities.

*Byproduct Material*.

1. Any radioactive material (except special nuclear material) yielded in, or made radioactive by, exposure to the radiation incident to the process of producing or using special nuclear material.
2. The tailings or wastes produced by the extraction or concentration of uranium or thorium from ore processed primarily for its source material content, including discrete surface wastes resulting from uranium solution extraction processes. Underground ore bodies depleted by these solution extraction operations do not constitute "byproduct material" within this definition.
3. (i) Any discrete source of radium (Ra)-226 that is produced, extracted, or converted after extraction, before, on, or after August 8, 2005, for use for a commercial, medical, or research activity; or
   (ii) Any material that—
      a. Has been made radioactive by use of a particle accelerator, and
      b. Is produced, extracted, or converted after extraction, before, on, or after August 8, 2005, for use for a commercial, medical, or research activity.
4. Any discrete source of naturally occurring radioactive material, other than source material, that—
   (i) The Commission, in consultation with the Administrator of the Environmental Protection Agency, the Secretary of Energy, the Secretary of Homeland Security, and the head of any other appropriate Federal agency, determines would pose a threat similar to the threat posed by a discrete source of Ra-226 to public health and safety or the common defense and security and
   (ii) Before, on, or after August 8, 2005, is extracted or converted after extraction for use in a commercial, medical, or research activity (see 10 CFR 20.1003).

*Categorical Exclusion*. A category of regulatory actions that do not individually or cumulatively have a significant effect on the human environment and that the Commission has found to have no such effect, in accordance with procedures set out in 10 CFR 51.22, "Criterion for Categorical Exclusion; Identification of Licensing and Regulatory Actions Eligible for Categorical Exclusion or Otherwise Not Requiring Environmental Review," and for which, therefore, neither an environmental assessment nor an environmental impact statement is required (see 10 CFR 51.14(a)).

*Characterization Survey*. A type of survey that includes facility or site sampling, monitoring, and analysis activities to determine the extent and nature of residual radioactivity. Characterization surveys provide the basis for acquiring necessary technical information to develop, analyze, and select appropriate cleanup techniques.

*Cleanup*. See *Decontamination.*

*Closeout Inspection*. An inspection performed by the NRC, or its contractor, to determine if a licensee has adequately decommissioned its facility. Typically, a closeout inspection is performed after the licensee has demonstrated that its facility is suitable for release in accordance with NRC requirements.

*Confirmatory Survey*. A survey conducted by the NRC, or its contractor, to verify the results of the licensee's final status survey. Typically, confirmatory surveys consist of measurements at a fraction of the locations previously surveyed by the licensee, to determine whether the licensee's results are valid and reproducible.

*Critical Group*. The group of individuals reasonably expected to receive the greatest exposure to residual radioactivity for any applicable set of circumstances (see 10 CFR 20.1003).

*DandD Code*. The Decontamination and Decommissioning (DandD) software package, developed by the NRC, that addresses compliance with the dose criteria of 10 CFR Part 20, "Standards for Protection against Radiation," Subpart E, "Radiological Criteria for License Termination." Specifically, DandD embodies the NRC's guidance on screening dose assessments to allow licensees to perform simple estimates of the annual dose from residual radioactivity in soils and on building surfaces. The current version of the code is 2.4, as of publication of this NUREG.

*Data Quality Objectives (DQOs).* Qualitative and quantitative statements derived from the *Data Quality Objectives process* that clarify study technical and *quality* objectives, define the appropriate type of data, and specify tolerable levels of potential decision errors that will be used as the basis for establishing the *quality* and quantity of data needed to support decisions.

*Data Quality Objectives Process (DQO process):* A series of logical steps that guides managers or staff to plan for the resource-effective acquisition of environmental data. See also *data quality objectives*.

*Decommission*. To remove a facility or site safely from service and reduce residual radioactivity to a level that permits (1) release of the property for unrestricted use and termination of the license or (2) release of the property under restricted conditions and termination of the license (see 10 CFR 20.1003).

*Decommissioning Groups*. For the purposes of this guidance document, the categories of decommissioning activities that depend on the type of operation and the residual radioactivity.

*Decommissioning Plan (DP)*. A detailed description of the activities that the licensee intends to use to assess the radiological status of its facility, to remove radioactivity attributable to licensed operations at its facility to levels that permit release of the site in accordance with the NRC's regulations and termination of the license, and to demonstrate that the facility meets the NRC's requirements for release. A DP typically consists of several interrelated components, including (1) site characterization information, (2) a remediation plan that has several components, including a description of remediation tasks, a health and safety plan, and a quality assurance (QA) plan, (3) site-specific cost estimates for the decommissioning, and (4) a final status survey plan (see 10 CFR 30.36(g)(4)).

*Decontamination*. The removal of undesired residual radioactivity from facilities, soils, or equipment before the release of a site or facility and termination of a license; also known as remediation, remedial action, and cleanup.

*Derived Concentration Guideline Levels (DCGLs)*. Radionuclide-specific concentration limits used by the licensee during decommissioning to achieve the regulatory dose standard that permits the release of the property and termination of the license. The DCGL applicable to the average concentration over a survey unit is called the $DCGL_W$. The DCGL applicable to limited areas of elevated concentrations within a survey unit is called the $DCGL_{EMC}$ (elevated measurement comparison).

*Distribution Coefficient or $K_d$.* Ratio of the concentration of an element or chemical associated with the soil to the concentration in the surrounding aqueous solution when the system is at equilibrium. The units are typically expressed in L/kg.

*Dose* (or *Radiation Dose*). A generic term that means absorbed dose, dose equivalent, effective dose equivalent, committed dose equivalent, committed effective dose equivalent, or total effective dose equivalent, as defined in 10 CFR 20.1003. In this report, dose generally refers to *total effective dose equivalent (TEDE)*.

*Durable Institutional Controls*. Durable institutional controls are reliable and sustainable for the time period needed. An institutional control that involves government ownership or control of the site would be considered a durable institutional control.

*Effluent*. Material discharged into the environment from licensed operations.

*Environmental Assessment*. A concise public document for which the Commission is responsible that serves to (1) briefly provide sufficient evidence and analysis for determining whether to prepare an environmental impact statement or a finding of no significant impact, (2) aid the Commission's compliance with the National Environmental Policy Act (NEPA) when no environmental impact statement is necessary, and (3) facilitate preparation of an environmental impact statement when one is necessary (see 10 CFR 51.14(a)).

*Environmental Impact Statement*. A detailed written document that ensures the NEPA policies and goals are considered in the actions of the Federal government. It discusses significant impacts and reasonable alternatives to the proposed action.

*Environmental Monitoring*. The process of sampling and analyzing environmental media in and around a facility (1) to confirm compliance with performance objectives and (2) to detect radioactive material entering the environment to facilitate timely remedial action.

*Environmental Report*. A document submitted to the NRC by an applicant for a license amendment request (see 10 CFR 51.14(a)). The NRC staff uses the environmental report to prepare environmental assessments and environmental impact statements. The requirements for environmental reports are specified in 10 CFR 51.45–69.

*Exposure Pathway*. The route by which radioactivity travels through the environment to eventually cause radiation exposure to a person or group.

*Exposure Scenario*. A description of the potential future land uses, human activities, and transport of radioactivity in the natural system as it influences a future human receptor's interaction with (and therefore exposure to) residual radioactivity. In particular, the exposure scenario describes where humans may be exposed to residual radioactivity in the environment, what exposure group habits determine exposure, and how residual radioactivity moves through the environment.

*External Dose*. That portion of the dose equivalent received from radiation sources outside the body (see 10 CFR 20.1003).

*Final Status Survey (FSS)*. Measurements and sampling to describe the radiological conditions of a site or facility, following completion of decontamination activities (if any) and in preparation for release of the site or facility.

*Final Status Survey Plan (FSSP)*. The description of the final status survey design.

*Final Status Survey Report (FSSR)*. The results of the final status survey conducted by a licensee to demonstrate the radiological status of its facility. The FSSR is submitted to the NRC for review and approval.

*Financial Assurance*. A guarantee or other financial arrangement provided by a licensee that funds for decommissioning will be available when needed. This is in addition to the licensee's regulatory obligation to decommission its facilities.

*Financial Assurance Mechanism*. Financial instruments used to provide financial assurance for decommissioning.

*Floodplain*. The lowland and relatively flat areas adjoining inland and coastal waters, including flood-prone areas of offshore islands. Areas subject to a one-percent or greater chance of flooding in any given year are included (see 10 CFR 72.3, “Definitions”).

*Footprint*. The portion of a site undergoing decommissioning, which comprises all the areas of soil containing residual radioactivity, where intentional mixing is proposed to meet the release criteria.

*General Licenses*. Licenses that are effective without filing applications with the NRC or the issuance of licensing documents to particular persons. The requirements for general licenses are found in 10 CFR Part 30, “Rules of General Applicability to Domestic Licensing of Byproduct Material;” 10 CFR Part 31, “General Domestic Licenses for Byproduct Material;” 10 CFR Part 40, “Domestic Licensing of Source Material;” and 10 CFR Part 70, “Domestic Licensing of Special Nuclear Material.” Examples of items for which general licenses are issued include tritium exit signs and anti-static devices.

*Groundwater*. Water contained in pores or fractures in either the unsaturated or saturated zones below ground level.

*Historical Site Assessment (HSA)*. The identification of potential, likely, or known sources of radioactive material and radioactive contamination based on existing or derived information for the purpose of classifying a facility or site, or parts thereof, as impacted or nonimpacted (see 10 CFR 50.2, “Definitions”).

*Hydraulic Conductivity.* The volume of water that will move through a medium in a unit of time under a unit hydraulic gradient through a unit area measured perpendicular to the direction of flow.

*Hydrology*. Study of the properties, distribution, and circulation of water on the surface of the land, in the soil and underlying rocks, and in the atmosphere.

*Impact*. The positive or negative effect of an action (past, present, or future) on the natural environment (land use, air quality, water resources, geological resources, ecological resources, aesthetic and scenic resources) and the human environment (infrastructure, economics, social, and cultural).

*Impacted Areas*. The areas with some reasonable potential for residual radioactivity in excess of natural background or fallout levels (see 10 CFR 50.2).

*Inactive Outdoor Area*. The outdoor portion of a site not used for licensed activities or materials for 24 months or more.

*Infiltration.* The process of water entering the soil at the ground surface. Infiltration becomes percolation when water has moved below the depth at which it can be removed (to return to the atmosphere) by evaporation or transpiration.

*Insignificant Radionuclides and Pathways*. Radionuclides and pathways that can be excluded from further detailed consideration, because they cumulatively contribute no more than 10 percent of the dose standard (e.g., for unrestricted release, no more than 0.025 mSv/y or 2.5 mrem/y). The dose contributions of the insignificant radionuclides and pathways should still be considered in demonstrating compliance with release criteria. See Chapter 3, Section 3.3 for additional information on insignificant radionuclides and pathways.

*In Situ Recovery (ISR).* In situ recovery (ISR) is one of the two primary extraction methods that are currently used to obtain uranium from underground. ISR facilities recover uranium from low-grade ores where other mining and milling methods may be too expensive or environmentally disruptive. The ISR process is as follows:

(1) A solution called lixiviant (typically containing water mixed with oxygen and/or hydrogen peroxide, as well as sodium carbonate or carbon dioxide) is injected through a series of wells into the ore body to dissolve the uranium.

(2) The lixiviant is then collected in a series of recovery wells, through which it is pumped to a processing plant, where the uranium is extracted from the solution through an ion-exchange process.

(3) The uranium extract is then further purified, concentrated, and dried to produce a material, which is called "yellowcake" because of its yellowish color.

(4) Finally, the yellowcake is packed in 55-gallon drums to be transported to a uranium conversion facility, where it is processed through the stages of the nuclear fuel cycle to produce fuel for use in nuclear power reactors.

*Institutional Controls*. Measures to control access to a site and minimize disturbances to engineered measures established by the licensee to control the residual radioactivity. Institutional controls include administrative mechanisms (e.g., land use restrictions) and may include, but are not limited to, physical controls (e.g., signs, markers, landscaping, and fences).

*Karst*. A type of topography that is formed over limestone, dolomite, or gypsum by dissolution, characterized by sinkholes, caves, and underground drainage.

*Leak Test*. A test for leakage of radioactivity from sealed radioactive sources. These tests are made when the sealed source is received and on a regular schedule thereafter. The frequency is usually specified in the sealed source and device registration certificate or license.

*License Termination Plan (LTP).* A detailed description of the activities a reactor licensee intends to use to assess the radiological status of its facility, to remove radioactivity attributable to licensed operations at its facility to levels that permit release of the site in accordance with the NRC's regulations and termination of the license, and to demonstrate that the facility meets the NRC's requirements for release. An LTP consists of several interrelated components, including

(1) a site characterization, (2) identification of remaining dismantlement activities, (3) plans for site remediation, (4) detailed plans for the final radiation survey, (5) a description of the end use of the facility, if restricted, (6) an updated site-specific estimate of remaining decommissioning costs, and (7) a supplement to the environmental report, pursuant to 10 CFR 51.33, "Draft Finding of No Significant Impact; Distribution," describing any new information or significant environmental change associated with the licensee's proposed termination activities (see 10 CFR 50.82, "Termination of License").

*License Termination Rule (LTR)*. The License Termination Rule refers to the final rule on "Radiological Criteria for License Termination," published by the NRC as Subpart E to 10 CFR Part 20 on July 21, 1997 (62 FR 39058).

*Licensee*. A person who possesses a license[1], or a person who possesses licensable material, whom the NRC could require to obtain a license.

*MARSSIM*. NUREG-1575, "Multi-Agency Radiation Site Survey and Investigation Manual," is a multiagency consensus manual that provides information on planning, conducting, evaluating, and documenting building surface and surface soil final status radiological surveys for demonstrating compliance with dose- or risk-based regulations or standards.

*Model.* A simplified representation of an object or natural phenomenon. The model can be in many possible forms, such as a set of equations or a physical, miniature version of an object or system constructed to allow estimates of the behavior of the actual object or phenomenon when the values of certain variables are changed. Important environmental models include those estimating the transport, dispersion, and fate of chemicals in the environment.

*Monitoring*. Monitoring (radiation monitoring, radiation protection monitoring) is the measurement of radiation levels, concentrations, or quantities of radioactive material and the use of the results of these measurements to evaluate potential exposures and doses (see 10 CFR 20.1003).

*mrem/y (millirem per year)*. One one-thousandth (0.001) of a rem per year. (See also *Sievert*.)

*National Environmental Policy Act (NEPA)*. The National Environmental Policy Act of 1969, which requires Federal agencies, as part of their decision-making process, to consider the environmental impacts of actions under their jurisdiction. Both the Council on Environmental Quality and the NRC have regulations to implement NEPA requirements. The Council's regulations are contained in 40 CFR Parts 1500 to 1508, and NRC requirements are provided in 10 CFR Part 51, "Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions."

*Naturally Occurring Radioactive Material (NORM)*. The natural radioactivity in rocks, soils, air, and water. NORM generally refers to materials in which the radionuclide concentrations have not been enhanced by or are as a result of human practices. NORM does not include uranium or thorium in source material.

*Non-impacted Areas*. The areas with no reasonable potential for residual radioactivity in excess of natural background or fallout levels (see 10 CFR 50.2).

[1] A license issued under the regulations in 10 CFR Parts 30 through 36, 39, 40, 50, 60, 61, 63, 70, or 72 (see definition in 10 CFR 20.1003).

*Pathway*. See *Exposure Pathway*.

*Performance-Based Approach*. Regulatory decision-making that relies upon measurable or calculable outcomes (i.e., performance results) to be met but provides more flexibility to the licensee as to the means of meeting those outcomes.

*Permeability.* The ability of a material to transmit fluid through its pores when subjected to a difference in head (pressure gradient). Permeability depends on the substance transmitted (e.g., oil, air, water) and on the size and shape of the pores, joints, and fractures in the medium and the manner in which they are interconnected.

*Porosity*. The ratio of openings, or voids, to the total volume of a soil or rock, expressed as a decimal fraction or as a percentage.

*Potentiometric Surface*. The two-dimensional surface that describes the elevation of the water table. In an unconfined aquifer, the potentiometric surface is at the top of the water level. In a confined aquifer, the potentiometric surface is above the top of the water level, because the water is under confining pressure.

*Principal Activities.* Activities authorized by the license that are essential to achieving the purpose(s) for which the license was issued or amended. Storage during which no licensed material is accessed for use or disposal and activities incidental to decontamination or decommissioning are not principal activities (see 10 CFR 30.4, “Definitions”).

*Probabilistic*. Refers to computer codes or analyses that use a random sampling method to select parameter values from a distribution. Results of the calculations are also in the form of a distribution of values. The results of the calculation do not typically include the probability of the exposure scenario occurring.

*Quality Assurance Project Plan (QAPP)*. A planning document that provides comprehensive details regarding the necessary quality assurance and quality control and other technical activities that must be implemented to ensure that the results of the work performed will satisfy stated performance criteria. See Appendix D of this volume for additional information on QAPPs.

*Reasonable Alternatives*. Those alternatives that are practical or feasible from a technical and economic standpoint.

*Reasonably Foreseeable Land Use*. Land use exposure scenarios that are likely within 100 years, considering advice from land use planners and stakeholders on land use plans and trends.

*Receptor Scenario.* See *Exposure Scenario.*

*rem*. The special unit of any of the quantities expressed as dose equivalent. The dose equivalent in rems is equal to the absorbed dose in rads multiplied by the quality factor (1 rem = 0.01 sievert) (see 10 CFR 20.1004, “Units of Radiation Dose”).

*Remedial Action*. See *Decontamination*.

*Remediation*. See *Decontamination*.

*Residual Radioactivity*. Radioactivity in structures, materials, soils, groundwater, and other media at a site resulting from activities under the licensee's control. This includes radioactivity from all licensed and unlicensed sources used by the licensee but excludes background radiation. It also includes radioactive materials remaining at the site as a result of routine or accidental releases of radioactive material at the site and previous burials at the site, even if those burials were made in accordance with the provisions of 10 CFR Part 20 (see 10 CFR 20.1003).

*RESRAD Code*. A computer code developed by the U.S. Department of Energy and designed to estimate radiation doses and risks from RESidual RADioactive materials in soils.

*RESRAD-BUILD Code*. A computer code developed by the U.S. Department of Energy and designed to estimate radiation doses and risks from RESidual RADioactive materials in BUILDings.

*RESRAD-OFFSITE Code.* A computer code developed by the U.S Department of Energy that extends the capabilities of the RESRAD (onsite) computer code to estimate the radiological consequences to a receptor located either onsite or outside the area of primary contamination.

*Restricted Area*. Any area to which access is limited by a licensee for the purpose of protecting individuals from undue risks from exposure to radiation and radioactive materials (see 10 CFR 20.1003).

*Risk*. Defined by the "risk triplet" of a scenario (a combination of events or conditions that could occur) or set of scenarios, the probability that the scenario could occur, and the consequence (e.g., dose to an individual) if the scenario were to occur.

*Risk-Based Approach*. Regulatory decision-making that is based solely on the numerical results of a risk assessment. (Note that the Commission does not endorse a risk-based regulatory approach.)

*Risk-Informed Approach*. Regulatory decision-making that represents a philosophy whereby risk insights are considered together with other factors to establish requirements that better focus licensee and regulatory attention on design and operational issues commensurate with their importance to public health and safety.

*Risk Insights*. Results and findings that come from risk assessments.

*Robust Engineered Barrier*. A man-made structure that is designed to mitigate the effect of natural processes or human uses that may initiate or accelerate the release of residual radioactivity through environmental pathways. The structure is designed so that the radiological criteria for license termination (10 CFR Part 20, Subpart E) are met. Robust engineered barriers are designed to be more substantial, reliable, and sustainable for the time period needed without reliance on active ongoing maintenance.

*Safety Evaluation Report*. The NRC staff's evaluation of the licensee's proposed action to determine if that action can be accomplished safely.

*Saturated Zone.* That part of the earth's crust beneath the regional water table in which all voids, large and small, are filled with water under pressure greater than atmospheric.

*Scoping Survey*. A type of survey that is conducted to identify (1) radionuclide contaminants, (2) relative radionuclide ratios, and (3) general levels and extent of residual radioactivity.

*Scenario.* As specified in draft low-level waste guidance (NUREG-2175, "Guidance for Conducting Technical Analyses for 10 CFR Part 61, Draft Report for Comment," issued March 2015), the term "scenario" refers to the expected ("central scenario") or potential ("alternative scenario") future dynamic evolution of the disposal site, which might include consideration of disruptive events (e.g., gully erosion, climate change). Typically, detailed consideration of the future evolution of a decommissioning site is unnecessary. However, central and alternative scenarios may need to be considered to demonstrate compliance with the radiological criteria for license termination for some complex decommissioning sites, or sites with relatively long-lived radionuclides (half-lives comparable to or longer than the compliance period). In general, the term "scenario" pertains to the "exposure scenario" (or receptor scenario), when used in NUREG-1757, Volume 2, as defined above.

*Screening Analysis/Approach/Methodology/Process*. The use of (1) predetermined building surface concentration and surface soil concentration values, or (2) a predetermined methodology (e.g., use of the DandD code) that meets the radiological decommissioning criteria without further analysis, to simplify decommissioning in cases where low levels of residual radioactivity are achievable.

*Sealed Source*. Any special nuclear material or byproduct material encased in a capsule designed to prevent leakage or escape of the material.

*sievert (Sv)*. The SI unit of any of the quantities expressed as dose equivalent. The dose equivalent in sieverts is equal to the absorbed dose in grays multiplied by the quality factor (1 sievert = 100 rem) (see 10 CFR 20.1004).

*Site*. The area of land, along with structures and other facilities, as described in the original NRC license application, plus any property outside the originally licensed boundary added for the purpose of receiving, possessing, or using radioactive material at any time during the term of the license, as well as any property where radioactive material was used or possessed that has been released before license termination.

*Site Characterization*. Studies that enable the licensee to sufficiently describe the conditions of the site, separate building, or outdoor area to evaluate the acceptability of the DP.

*Site Characterization Survey.* See *Characterization Survey*.

*Site Decommissioning Management Plan (SDMP)*. The program established by the NRC in March 1990 to help ensure the timely cleanup of sites with limited progress in completing the remediation of the site and the termination of the facility license. SDMP sites typically have buildings, former waste disposal areas, large volumes of tailings, groundwater contamination, and soil contaminated with low levels of uranium or thorium or other radionuclides.

*Site-Specific Dose Analysis*. Any dose analysis that is performed other than by using the default screening tools.

*Smear*. A radiation survey technique that is used to determine levels of removable surface contamination. A medium (typically filter paper) is rubbed over a surface (typically of an area

100 square centimeters), followed by a quantification of the activity on the medium. Also known as a swipe.

*Source Material*. Uranium or thorium, or any combination of uranium and thorium, in any physical or chemical form, or ores that contain by weight one-twentieth of one percent (0.05 percent) or more of uranium, thorium, or any combination of uranium and thorium. Source material does not include special nuclear material (see 10 CFR 20.1003).

*Source Term*. Release rates associated with residual radioactivity at a site or facility. The source term is related to the inventory, distribution of contamination, and controlling release mechanisms (e.g., solubility-controlled release, diffusion-limited release, or desorption). Note that the working definition of source term in this volume is slightly different than the definition of source term found in the NRC glossary. The definition found in the online NRC glossary, at https://www.nrc.gov/reading-rm/basic-ref/glossary/source-term.html, is specific to accidents involving radioactive materials: "types and amounts of radioactive or hazardous material released to the environment following an accident".

*Special Nuclear Material*. (1) Plutonium, uranium (U)-233, uranium enriched in the isotope 233 or in the isotope 235, and any other material that the Commission, pursuant to the provisions of Section 51 of the Atomic Energy Act of 1954, as amended (AEA), determines to be special nuclear material, but does not include source material; or (2) any material artificially enriched by any of the foregoing but does not include source material (see 10 CFR 20.1003).

*Specific Licenses*. Licenses issued to a named person who has filed an application for the license under the provisions of 10 CFR Parts 30, 32–36, 39, 40, 61, 70, and 72. Examples of specific licenses are industrial radiography, medical use, irradiators, and well logging.

*Surface Soil*. The top layer of soil on a site that supports certain exposure pathways such as direct exposure, soil ingestion, and resuspension of particles for inhalation. Surface soil has also been associated with the thickness of soil that can be measured using direct measurement or scanning techniques. Typically, this layer is often represented as the top 15 centimeters (6 inches) of soil but will vary depending on the radionuclide, surface characteristics, measurement technique, and dose modeling assumptions.

*Survey*. An evaluation of the radiological conditions and potential hazards incident to the production, use, transfer, release, disposal, or presence of radioactive material or other sources of radiation. When appropriate, such an evaluation includes a physical survey of the location of radioactive material and measurements or calculations of levels of radiation, or concentrations or quantities of radioactive material present (see 10 CFR 20.1003).

*Survey Unit*. A geographical area consisting of structures or land areas of specified size and shape at a site for which a separate decision will be made as to whether the unit attains the site-specific reference-based cleanup standard for the designated pollution parameter. Survey units are generally formed by grouping contiguous site areas with similar use histories and having the same contamination potential (classification). Survey units are established to facilitate the survey process and the statistical analysis of survey data.

*Timeliness*. Specific time periods stated in NRC regulations for decommissioning unused portions of operating nuclear materials facilities and for decommissioning the entire site upon termination of operations.

*Total Effective Dose Equivalent (TEDE)*. The sum of the deep-dose equivalent (for external exposures) and the committed effective dose equivalent (for internal exposures) (see 10 CFR 20.1003).

*Transmissivity.* The rate of flow of water through a vertical strip of aquifer that is one unit wide and that extends the full saturated depth of the aquifer.

*Unrestricted Area.* An area, access to which is neither limited nor controlled by the licensee (see 10 CFR 20.1003).

*Unsaturated Zone.* The subsurface zone in which the geological material contains both water and air in pore spaces. The top of the unsaturated zone typically is at the land surface, otherwise known as the vadose zone.

*Vadose Zone*. See *Unsaturated Zone*.

# 1 PURPOSE, APPLICABILITY, AND ROADMAP

## 1.1 Purpose and Applicability of this Volume

The purpose of this volume is to do the following:

- Provide guidance to NRC licensees for demonstrating compliance with the radiological criteria for license termination. Specifically, provide guidance relevant to demonstrating compliance with 10 CFR Part 20, Subpart E, for materials and reactor licensees.
- Provide guidance to the NRC staff on methods and techniques acceptable for compliance with the license termination criteria.
- Maintain a risk-informed, performance-based, and flexible decommissioning approach.

This NUREG provides guidance on decommissioning leading to termination of a license. Licensees decommissioning their facilities are required to demonstrate to the NRC that their proposed methods will ensure that the decommissioning can be conducted safely and that the facility, at the completion of decommissioning activities, will comply with the NRC's requirements for license termination. This volume is also intended to be used in conjunction with NRC Inspection Manual Chapter 2602, "Decommissioning Oversight and Inspection Program for Fuel Cycle Facilities and Materials Licensees." Licensees who are subject to Subpart E, "Radiological Criteria for License Termination," of 10 CFR Part 20, "Standards for Protection against Radiation," should use the policies and procedures discussed in this volume to develop and implement a decommissioning plan (DP) or license termination plan (LTP) (note that throughout this volume, when the term "DP" is used, it may generally be understood to refer to DPs or LTPs). Uranium recovery facilities may find this information useful, but they are not subject to Subpart E. Agreement State licensees should contact the appropriate regulatory authority. Depending on the State, Agreement State licensees may be able to use this guidance with the substitution of "Agreement State Authority" for "NRC."

Additionally, there are several military and former military sites around the country where the responsible Federal agency (e.g., U.S. Department of the Air Force, U.S. Department of the Army, or U.S. Department of the Navy) is implementing site reclamation activities to address the removal or remediation of radiological material under the Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA), as amended (42 U.S.C. §§ 9601 et seq.), also known as Superfund. The NRC has received a number of inquiries into its regulatory jurisdiction at various sites where the responsible Federal agency uses non-Federal entities (i.e., private service providers) to conduct remediation activities involving regulated radioactive materials (byproduct, source, or special nuclear materials) (see Atomic Energy Act of 1954, as amended (AEA), 42 U.S.C. §§ 2011–2297h (2006)) on Federal property located in an Agreement State. To assist it in making future determinations, the NRC staff has developed a decision process that is consistent with the procedures in SA-500, "Jurisdiction Determinations," dated March 10, 2011; the decision process is found in FSME-14-039, "Clarification on the Determination of Regulatory Jurisdiction of Non-Federal Entities Conducting Cleanup activities on Federal Property in Agreement States," dated April 22, 2014.

This volume of NUREG-1757 is being issued to describe and make available to licensees and the public (1) guidance on technical aspects of compliance with specific parts of the Commission's regulations, (2) methods acceptable to the NRC staff in implementing these

regulations, and (3) some of the techniques and criteria the NRC staff uses in evaluating DPs and LTPs. Licensees should use this guidance to prepare DPs, LTPs, final status surveys (FSSs), and other technical decommissioning reports for NRC submittal. The NRC staff will use the guidance in reviewing these documents and related license amendment requests. The guidance in this volume is not a substitute for regulations, and compliance with the guidance is not required. Methods and solutions different from those described in this volume will be acceptable, if licensees provide a sufficient basis for the NRC staff to conclude that the licensees' decommissioning actions are in compliance with the Commission's regulations. However, the use of nonstandard methods may require more detailed justification for the NRC staff to determine acceptability. In addition, the increased complexity and detail of nonstandard demonstrations may result in increased NRC staff review time and, therefore, cost to the licensee.

Volume 2 does not address the following:

- financial assurance for decommissioning
- public notification and participation
- recordkeeping and timeliness in decommissioning
- decommissioning of uranium recovery facilities
- disposition of solid materials from licensee control[1]

## 1.2 Roadmap to this Volume

The NRC's regulations require a licensee to submit a DP to support the decommissioning of its facility either (1) when it is required by a license condition or (2) when the NRC has not approved the procedures and activities necessary to carry out the decommissioning, and these procedures could increase the potential health and safety impact to the workers or the public. Chapters 4–6 provide acceptance criteria and evaluation criteria for use in reviewing DPs and other information submitted by licensees to demonstrate that the facility is suitable for release in accordance with NRC requirements.

The approach used in this volume is similar to that in Volume 1 of this NUREG report. Volume 1 described the categorization of facilities into Decommissioning Groups 1–7, based on the amount of residual radioactivity, the location of that material, and the complexity of the activities needed to decommission the site. Table 1.1 provides a summary description and examples of each decommissioning group (see Part I of Volume 1 of this NUREG series for more details). Table 1.2 shows the potential applicability of the guidance in this volume to each of these groups. Therefore, where possible, the guidance in this volume has been categorized by the decommissioning groups. For most topics in this volume, the guidance applies to more than one decommissioning group, as shown in Table 1.2. Licensees are encouraged to contact the appropriate NRC staff to determine the applicability of the guidance to their facility.

[1] Although dose modeling guidance in this NUREG volume may be useful for assessing dose to members of the public from the release of solid materials, unique scenarios and pathways specific to the release of solid materials are not within the scope of this guidance document.

**Table 1.1 Description and Examples of Each Decommissioning Group**

| Group | General Description | Typical Examples |
|---|---|---|
| 1 | Licensed material was not released into the environment, did not cause the activation of adjacent materials, and did not contaminate work areas. | Licensees who used only sealed sources, such as radiographers and irradiators |
| 2 | Licensed material was used in a way that resulted in residual radioactivity on building surfaces and/or soils. The licensee is able to demonstrate that the site meets the screening criteria for unrestricted use. | Licensees who used only quantities of loose radioactive material that they routinely cleaned up (e.g., research and development facilities) |
| 3 | Licensed material was used in a way that could meet the screening criteria, but the license needs to be amended to modify or add procedures to remediate buildings or sites. | Licensees who may have occasionally released radioactivity within NRC limits (e.g., broad scope) |
| 4 | Licensed material was used in a way that resulted in residual radiological contamination of building surfaces or soils, or a combination of both (but not groundwater). The licensee demonstrates that the site meets unrestricted use levels derived from site-specific dose modeling. | Licensees whose sites released loose or dissolved radioactive material within NRC limits and may have had some operational occurrences that resulted in releases above NRC limits (e.g., waste processors) |
| 5 | Licensed material was used in a way that resulted in residual radiological contamination of building surfaces, soils, or groundwater. The licensee demonstrates that the site meets unrestricted use levels derived from site-specific dose modeling. | Licensees whose sites released, stored, or disposed of large amounts of loose or dissolved radioactive material onsite (e.g., fuel cycle facilities) |
| 6 | Licensed material was used in a way that resulted in residual radiological contamination of building surfaces, soils, or groundwater. The licensee demonstrates that the site meets **restricted** use levels derived from site-specific dose modeling. | Licensees for whom cleaning their site to the **unrestricted** release limit would cause a greater health and safety or environmental impact than could be justified (e.g., facilities where large inadvertent release(s) occurred) |
| 7 | Licensed material was used in a way that resulted in residual radiological contamination of building surfaces, soils, or groundwater. The licensee demonstrates that the site meets **alternate restricted** use levels derived from site-specific dose modeling. | Licensees for whom cleaning their site to the **restricted** release limit would cause a greater health and safety or environmental impact than could be justified (e.g., facilities where large inadvertent release(s) occurred) |

**Table 1.2 Applicability of Volume 2 to Decommissioning Groups**

| Groups | Group 1 | Group 2 | Group 3 | Group 4 | Group 5 | Group 6 | Group 7 |
|---|---|---|---|---|---|---|---|
| **Dose Assessment Method** | N/A | Screening criteria (Section 5, Appendix H) | | Site-specific assessment (Section 5, Appendices I, J, M, and Q) | | | |
| **Dose Assessment for Partial Site Release** | No | Yes, for licensees electing partial site releases (Appendices K and L) | | | | | |
| **Site Characterization** | No | Yes (Section 4.2, Appendix E) | | | Yes (Section 4.2, Appendices E, F, G, and O) | | |
| **Remedial Action Support Surveys** | No | Yes, if remediation is required (Section 4.3, Appendix E) | | | | | |
| **Final Status Survey** | No | Yes (Sections 4.4 and 4.5, Appendices A, B, D, and E) | | Yes (Sections 4.4 and 4.5, Appendices A, B, D, and E) | | | |
| **Complex Survey Situations (Not Addressed in MARSSIM)** | No | | | Yes (Section 4.6, Appendices C, G and O) | | | |
| **Surface and Groundwater Characterization** | No | | | Yes[2] (Appendix F) | Yes (Appendix F) | | |
| **ALARA Analysis** | No | Yes, good housekeeping only (Section 6.3) | | Yes (Chapter 6, Appendix N) | | | |

2 Group 4 sites with residual radioactivity in surface water will benefit from Appendix A on surface water characterization.

Because of the variability in the amounts, forms, and types of radioactive material used by each decommissioning group, licensees may need to submit a broad range of information types and details to the NRC for approval of decommissioning activities. The types of information required could vary because of the radionuclides involved, whether remediation is required, or the complexity of the site. The amount of detail discussed in this volume is based on the needs of complicated sites. The NRC staff does not suggest that all licensees should provide the same level of detail. Rather, the amount of detail provided on a specific issue should be commensurate with the complexity of the issue for the facility. Thus, licensees and NRC reviewers should generally determine the level of detail and appropriate methods, based on the complexity of the facility as related to a compliance demonstration. Licensees are encouraged to discuss with their NRC license reviewer the appropriate level of detail to be included in the DP, using the checklists of Appendix D of Volume 1 of this NUREG report.

The technical aspects of sites, as related to decommissioning, are often called either “simple” or “complex.” The question becomes what defines the technical aspects as “simple” or “complex.” One needs to decide what aspect of the decommissioning one is trying to judge. For example, site characterization may be complex at a site, but the FSS, after remediation, may be simple and straightforward.

Unfortunately, there is no precise definition or list of characteristics that can define the technical aspects as either simple or complex without caveats. That is because simple and complex are not distinct boxes but part of a continuum. For example, sites using screening criteria are relatively “simple,” technically, and sites proposing both partial release and restricted release with an engineered barrier design along with institutional controls that rely on active maintenance are relatively “complex,” technically. While there can be exceptions to the site complexity characterization illustrated in Figure 1.1, Decommissioning Groups 1–3 generally have mostly simple technical aspects, and Decommissioning Groups 5–7 generally have mostly “complex” technical aspects. Group 4 sites, which are sites without initial groundwater contamination, can be of either complexity.


**Figure 1.1 Continuum of Site Complexity**

Simple sites are generally easy to assess, because site characterization information, survey methods, and models with NRC-reviewed default parameter sets are readily available. These sites have residual radioactivity generally limited to building surfaces or surface soil at a site with simple geological and hydrological characteristics.

Technically complex sites are generally sites with one or more of the following conditions:

- existing groundwater or surface water contamination
- former burials of radioactive material or highly heterogeneous subsurface soil residual radioactivity

- diversified and extensive surface or subsurface residual radioactivity that, because of the interactions between sources, may require data and modeling of these multiple sources at the site
- radionuclides that (1) are hard-to-detect (HTD), (2) lack suitable surrogate radionuclides, or (3) have very low effective derived concentration guideline levels (DCGLs)
- current offsite releases such that alternate offsite exposure scenario(s) may be required or the use of an onsite resident farmer exposure scenario may be inadequate (e.g., sites with multiple receptors)
- planned license termination under restricted conditions (10 CFR 20.1403, "Criteria for License Termination under Restricted Conditions")
- physical barriers or vaults
- unusual physical or lithologic properties, such as a highly fractured formation, karst features, or sinkholes that may significantly affect assumptions of transport models or the overall conceptual model

These conditions are not rigid definitions, as other factors are also important. One such important factor would be the locations where radionuclides are present. For example, a site could be called simple because the predominant radionuclide is a short-lived energetic gamma that is in the surface soil; even if the hydrology at the site is complex, the site would be called simple, because the primary exposure pathway is external exposure, which is an uncomplicated pathway.

Technically complex sites may require more advanced remediation, survey planning, or performance assessment modeling and analysis approaches. Specifically, more advanced approaches may be required to select appropriate models or codes, collect characterization data, justify source term assumptions, ensure internal consistencies in the associated complex transport models, and design site- or source-specific survey plans. Because of the complex nature of these sites, the scope of NRC staff review will depend on site-specific conditions and on the degree of site complexity. Therefore, a generic NRC staff review of complex sites cannot be articulated in this volume.

Licensees and the NRC staff should interact early for information and direction on the development of a complete DP. Once the decision has been made to decommission, the next step is to determine what information the licensee needs to demonstrate site conditions successfully. If the submittal of a DP is not necessary, the licensee should follow the guidance in Volume 1 of this NUREG report for the appropriate decommissioning group.

If the licensee is required to submit a DP, it should schedule a meeting with the NRC staff to discuss both the planned decommissioning and the approach that will be used to evaluate the information submitted to support the decommissioning. The NRC staff and the licensee should review the licensed operations, types and quantities of radioactive materials used at the facility, and any other activities (e.g., spills, leaks) that could affect decommissioning operations. The NRC staff should also discuss the decommissioning goal envisioned by the licensee (i.e., license termination under unrestricted versus restricted conditions) and the information required to be submitted for the appropriate decommissioning group (described in Chapters 10, 11, 12, 13, or 14 of Volume 1 of this NUREG report). The NRC staff should then discuss the acceptance criteria for information to be included in the DP. Finally, the NRC staff should prepare a site-specific checklist for evaluating the DP. Appendix D of Volume 1 of this NUREG report provides a generic checklist that may be used to develop this site-specific checklist.

Thus, before the licensee begins to develop its DP, both the NRC staff and the licensee should have a good understanding of the types of information that should be included in the DP, as well as the criteria that the NRC will use to evaluate the information submitted to support decommissioning. This should help minimize the need for requests for additional information.

## 1.3 Roadmap for Guidance on Restricted Use, Alternate Criteria, and Use of Engineered Barriers

The focus of this volume is on guidance for demonstrating compliance with the dose criteria from 10 CFR Part 20, Subpart E. However, there are additional criteria in Subpart E related to license termination under restricted conditions and the use of alternate criteria for license termination. In addition, some licensees may wish to use engineered barriers as part of the compliance strategy. This section describes where guidance on these subjects may be found in this NUREG series (Volumes 1 and 2).

Table 1.3 provides cross-references to sections of Volume 1 and this volume for guidance on aspects of restricted use, use of alternate criteria, and use of engineered barriers.

## 1.4 Iterative Nature of the Compliance Demonstration Process: A Decision-making Framework

The NRC staff developed an overall framework for dose assessment and decision-making where the licensee has decided to begin the decommissioning and license termination process at sites ranging from simple ones to the more complex or contaminated sites. This volume summarizes information for using the framework to step through the decommissioning and license termination process; NUREG-1549, "Decision Methods for Dose Assessment to Comply with Radiological Criteria for License Termination," issued July 1998, contains a detailed description. Although the framework was developed for demonstrating compliance using the characterization and dose assessment approach (see Section 2.5), the concepts may be extended for use in the DCGL development and the FSS approach.

This framework is designed to assist the licensee, the NRC staff, and other stakeholders in making decommissioning decisions. By doing so, the process allows the licensee to do the following:

- coordinate its planning efforts with NRC staff input and conduct dose assessments and site characterization activities that are directly related to regulatory decisions
- optimize cost decisions related to site characterization, remediation, and land use restrictions
- integrate analyses for requirements that are as low as is reasonably achievable (ALARA)
- elicit other stakeholder input at crucial points

**Table 1.3 Cross-References for Restricted Use, Alternate Criteria, and Use of Engineered Barriers**

| Issue | Applicable Sections of This Report | |
|---|---|---|
| | Volume 1 | Volume 2 |
| Initial eligibility demonstration for restricted use | 17.7.2 | n/a |
| Institutional controls | 17.7.3 | n/a |
| Site maintenance and long-term monitoring | 17.7.4 | n/a |
| Obtaining public advice | 17.7.5 and Appendix M | n/a |
| Dose modeling for restricted use | 17.7.6 | 5 |
| ALARA analysis for restricted use | 17.7.3.5 | 6 and Appendix N |
| Use of alternate criteria | 17.8 | n/a |
| Dose modeling for alternate criteria | 17.8 | 5 |
| Use of engineered barriers | 17.7.3 | 3.5 and Appendix P |

The framework is designed to allow the licensee flexibility in the decision-making process for demonstrating compliance. As such, the framework provides one method that may be useful for licensees in developing the compliance strategy.

The steps and decision points of the decision framework support an assessment of the entire range of dose modeling options from which a licensee may choose, whether it involves using generic screening parameters, changing parameters, or modifying pathways or models. Figure 1.2 (modified from NUREG-1549) illustrates the decision framework, including its steps and decision points.

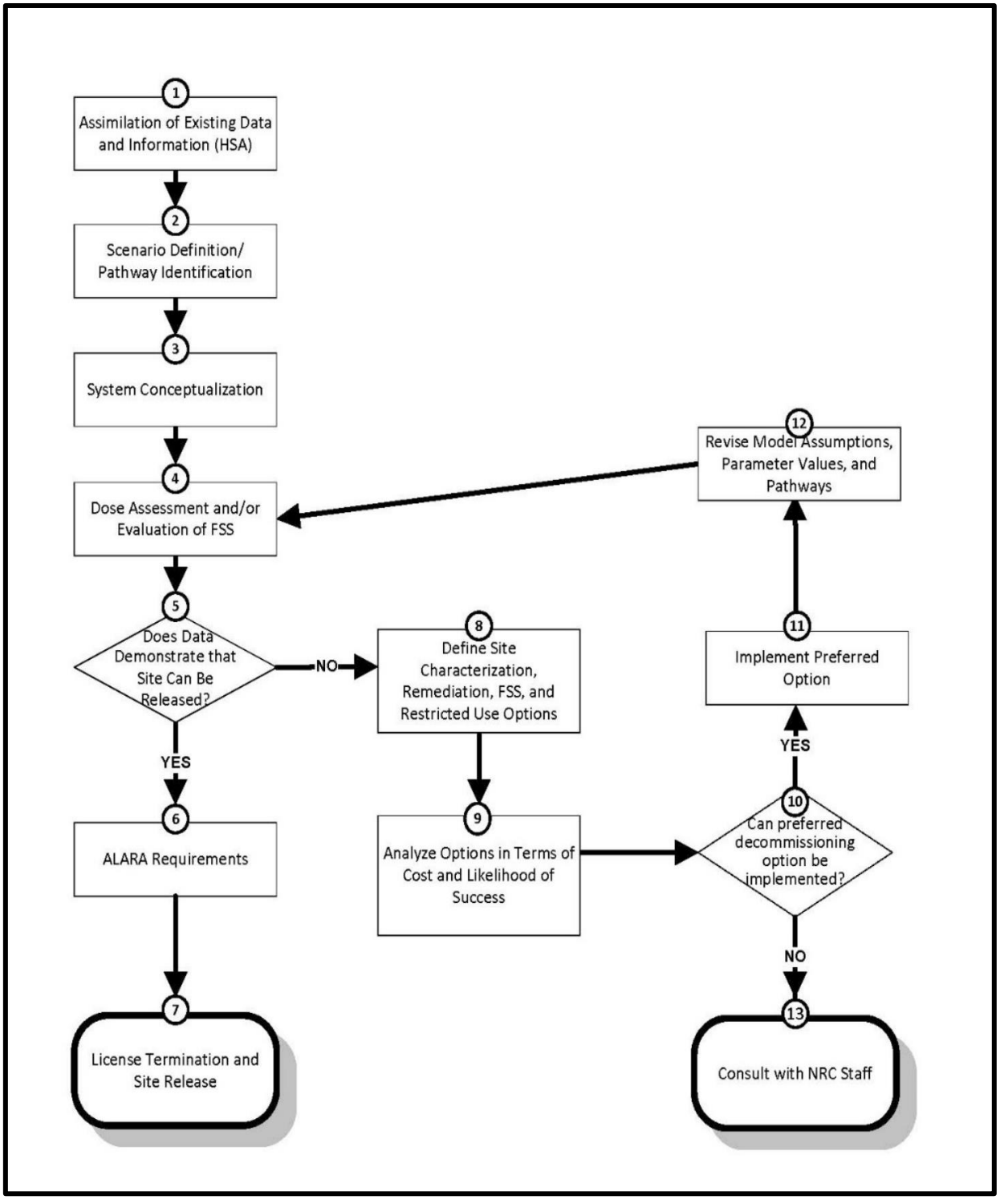


**Figure 1.2 Decommissioning and License Termination Decision Framework (Modified from NUREG-1549)**

### 1.4.1 Contents and General Concepts of the Iterative Approach in Using the Decision Framework

To facilitate the preparation and evaluation of the dose assessments, this framework describes an iterative approach to decision-making for license termination. An iterative approach is helpful because of the very wide range of levels of residual radioactivity, complexity of analysis, and potential remediation necessary at NRC-licensed sites. The iterative approach consists of using existing information for generic screening and using site-specific information as appropriate. This approach provides assurance that obtaining additional site-specific information is worthwhile, because it ensures that a more realistic dose assessment will generally result in an estimated dose no greater than that estimated using screening. These two phases of the compliance assessment are summarized in broad terms below, while NUREG-1549 contains further details:

(1) Generic screening: In this iteration, licensees would demonstrate compliance with the dose criteria of the License Termination Rule (LTR) by using predefined models and generic screening parameters.

(2) Use of site-specific information as appropriate: If compliance cannot be demonstrated using generic screening, then licensees should proceed to the next iteration of analysis in which defensible site-specific values are obtained and applied.

The following general concepts apply to using the iterative approach with the decision framework shown in Figure 1.2:

- The approach provides a process for screening sites and for directing additional data collection efforts where necessary or where it is most helpful toward demonstrating compliance.

- The framework is designed such that the level of complexity and rigor of analysis conducted for a given site should be commensurate with the level of risk that the site poses.

- The licensee would not need to start the process with generic screening but may move directly to the use of site-specific information, as appropriate.

- For the process to work efficiently, the licensee is encouraged to involve the NRC staff from the very first step through to the end of the decision-making process.

The framework provides the licensee with a variety of options for performing dose assessments, from simple screening to more detailed site-specific analyses. Use of the framework would normally encompass Steps 1–7; however, the amount of work that goes into each of these steps should be based on the expected levels of residual radioactivity and the health risks they pose. Note that, in this framework, while all sites may start at the same level of very simple analyses (not a requirement for successful implementation), it is expected that only certain sites would progress to very complex dose assessment and options analyses. Some sites may not need to conduct any options analyses, as described in Step 8, and some sites may need to evaluate a limited set of relatively simple and inexpensive options. For example, the licensee at a site with a contained source of residual radioactivity that is obviously simple to remove would not spend time analyzing large suites of alternative data collection and remediation options. On the other hand, the licensee at a site with high levels of widely distributed residual radioactivity

may use this process to analyze a variety of simple and complex options to define the best decontamination and decommissioning strategy. Therefore, this approach ensures that the licensee's efforts and expenses will be commensurate with the level of risk posed by the site.

### 1.4.2 Steps of the Decision Framework

NUREG-1549 provides three separate discussions to illustrate the iterative nature of assessments as site complexity increases. The following is both a summary of the steps of the decision framework and a set of examples to help users understand most of the features of dose modeling in the context of the decision framework. This discussion has been modified slightly from that in NUREG-1549 to make it applicable to a broader range of compliance demonstrations. A number of the examples refer to the use of the Decontamination and Decommissioning (DandD) and RESidual RADioactive (RESRAD) materials dose assessment codes. Chapter 5 and Appendices H and I of this volume discuss dose modeling codes, including specifics on these two dose assessment codes, while NUREG-1549 contains further details. Figure 1.2 (modified from NUREG-1549) shows the following steps of the dose modeling framework:

(1) The first step in a compliance assessment involves gathering and evaluating existing data and information about the site for the historical site assessment (HSA), including the nature and extent of residual radioactivity at the site. Often, minimal information is all that is needed for an initial screening analysis (e.g., a simple representation of the source of residual radioactivity). Specifically, information is needed to support the decision that the site is "simple" and is qualified for a screening analysis. However, licensees should use all readily available information about the site. This step also includes the definition of the performance objectives for compliance with decommissioning criteria.

(2) This step involves defining the exposure scenarios and pathways that are important and relevant for the site dose assessment. This step also includes the preliminary determination of whether the licensee plans to adopt an unrestricted use or restricted use option provided for in the LTR. For all assessments using screening concentration tables or DandD, the NRC has already defined the generic exposure scenarios and pathways for screening. For a site-specific analysis, DandD and the RESRAD family of codes (e.g., RESRAD-ONSITE, RESRAD-OFFSITE, RESRAD-BUILD) may be used, in addition to other codes. The codes used should allow the licensee to select and deselect exposure pathways as appropriate for the site-specific conditions.

(3) Once exposure scenarios are defined and exposure pathways identified, the licensee develops a basic conceptual understanding of the system, often based on simplifying assumptions on the nature and behavior of the natural systems. System conceptualization includes conceptual and mathematical model development and an assessment of parameter uncertainty. Using DandD for generic screening (and as the basis for screening concentration tables), the NRC has predefined conceptual models for the exposure scenarios, along with default parameter distributions (based on NUREG/CR-5512, "Residual Radiation Contamination from Decommissioning," Volumes 1 and 3, issued October 1992). The site-specific analysis can use DandD or the RESRAD family of codes, after verification that the site conceptual model is compatible with the conceptual model of the code selected.

(4) This step involves the dose assessment or consequence analysis, based on the defined exposure scenario(s), exposure pathways, models, and parameter distributions. This step may also involve the evaluation of FSS results. For generic screening, reviewers can accept lookup tables and use the generic models and default parameter probability density functions (PDFs) by running DandD with the appropriate site-specific source concentrations and configuration, while leaving all other information in the software unchanged. Site-specific assessments allow the licensee to use other codes and change pathways and parameter distributions based on site-specific data and information. DandD, and the RESRAD family of codes provide various plots and reports of the dose distribution using Monte Carlo sampling of the input distributions.

(5) This is the first major decision point in the license termination decision process. It involves answering the question of whether the dose assessment results and/or FSS results demonstrate compliance with the dose criteria in 10 CFR Part 20, Subpart E. If the results demonstrate compliance, the licensee proceeds with Steps 6 and 7 to meet the ALARA requirements in Subpart E. If the results are ambiguous or clearly exceed the performance objective, then the licensee proceeds to Steps 8 and 9 for the next iteration of the decision-making process.

(6) In this step, the licensee addresses the ALARA criterion of 10 CFR Part 20, Subpart E, if it has not already done so. If the ALARA requirements are satisfied, then the licensee initiates the license termination. Note that the DandD and the RESRAD family of codes do not involve or automate these steps.

(7) This step includes the administrative and other actions necessary to terminate the license and release the site, with Volume 1 of this NUREG containing more details on the specific actions to do so.

(8) Full application of the decision framework involves defining all possible options the licensee might address to defend a final set of actions needed to demonstrate compliance with license termination criteria. Options may include (i) acquiring more data and information about the site and source(s) of residual radioactivity to reduce uncertainty about the pathways, models, and parameters, and thus reduce the calculated dose, (ii) reducing actual contamination through remediation actions, (iii) reducing exposure to radionuclides through implementation of land use restrictions, (iv) performing an FSS, or (v) some combination of these options.

(9) All the options identified in Step 8 are analyzed and compared to optimize the selection of a preferred set of options. This options analysis may consider the cost of implementation, the likelihood of success (and the expected costs associated with success or failure to achieve the desired results when the option is implemented), the timing considerations and constraints, and other quantitative or qualitative selection criteria.

(10) The activities in Steps 8 and 9 provide information for licensees to choose the preferred decommissioning option based on cost, the likelihood of success, timeliness, and other considerations. For example, results of sensitivity analyses performed with DandD, or the RESRAD family of codes, can be used by a licensee to identify one or more parameters that may be modified, based on the acquisition of site-specific information and data. If new data can reduce the uncertainty associated with

parameters found to be important to dose, then the licensee may be able to defend a new calculated dose that meets the license termination criteria. This step may include submitting a DP to the NRC, if necessary, to proceed with the preferred option. If the licensee believes that no viable options exist at this time, the licensee should confer with the NRC staff (see also Step 13).

(11) Under this step, the preferred option is implemented. The licensee obtains the information necessary to support revisions to the parameters identified in Steps 8 and 9 or performs an FSS.

(12) Once the licensee obtains the data, it revises the affected parameters for the predefined models, as appropriate. Also, data may support the elimination of one or more of the exposure pathways in the predefined exposure scenarios. DandD and the RESRAD family of codes provide very simple and straightforward modifications of the pathways and parameters of interest.

Once the pathways and parameters are revised, the licensee would revisit Steps 4 and 5 to determine the impact of the revisions on demonstrating compliance with the performance objectives. If met, the licensee proceeds to Steps 6 and 7. If the performance objective is still exceeded, the licensee returns to Steps 8 and 9 to analyze the remaining options.

In certain limited circumstances, terminating the license may not be feasible. The licensee should contact the NRC staff for case-specific guidance and for the regulatory approvals that may be necessary to maintain, rather than terminate, the license.

# 2 FLEXIBILITY IN DEMONSTRATING COMPLIANCE WITH 10 CFR PART 20, SUBPART E

The NRC and its licensees share a common responsibility to protect public health and safety. Federal regulations and the NRC regulatory program are important elements in the protection of the public; however, NRC licensees are primarily responsible for safely using nuclear materials. The agency's safety philosophy explains that "although NRC develops and enforces the standards governing the use of nuclear installations and materials, it is the licensee who bears the primary responsibility for conducting those activities safely." This philosophy applies to the decommissioning of licensed facilities. Thus, the licensee has the primary responsibility for compliance with the license termination criteria. The responsibility of the NRC staff is to oversee the process and conclude that there is reasonable assurance that the criteria have been or will be met and then to terminate or amend licenses, as appropriate.

The dose criteria of 10 CFR Part 20, Subpart E, are performance criteria. In this volume, the NRC staff has taken a risk-informed, performance-based approach to demonstrations of compliance with the license termination criteria, using various methods available to licensees. Regardless of the specific method used, it is important that the licensee provide sufficient justification for its approach. This chapter discusses some of the aspects of flexibility in methodologies for demonstrating compliance with the license termination criteria. One objective of this chapter is to emphasize the flexibility available in demonstrating compliance with the regulations.

> Licensees should consider the flexibility available when demonstrating compliance with the license termination criteria. A licensee may determine that the standard methods are not the best for a given site. The benefit of the performance criteria is the flexibility of approaches allowed to demonstrate compliance.
>
> The NRC staff should evaluate any methodology proposed by licensees. However, the use of nonstandard methods may require more detailed justification for the NRC staff to determine acceptability. In addition, the increased complexity and detail of nonstandard demonstrations may result in increased NRC staff review time and, therefore, cost to the licensee.

## 2.1 Risk-informed Approach to Compliance Demonstrations and Reviews

This section summarizes the risk-informed approach to regulatory decision-making. The NRC staff requirements memorandum (SRM) for SECY-98-144, "White Paper on Risk-Informed and Performance-Based Regulation," issued March 1999, contains additional details.

The NRC has increased the use of risk information and insights in its regulation of nuclear materials and nuclear waste management, including the decommissioning of nuclear facilities. Risk is defined by the "risk triplet" of (1) either a scenario or set of scenarios with a combination of events and/or conditions that could occur, (2) the probability that the scenario(s) could occur, and (3) the consequence (e.g., the dose to an individual) if the scenario(s) were to occur. The term risk insights, as used here, refers to the results and findings that come from risk assessments. The end results of such assessments may relate directly or indirectly to public health effects (e.g., the calculation of predicted doses from decommissioned sites).

A risk-based approach to regulatory decision-making is based solely on the numerical results of a risk assessment. The Commission does not endorse a risk-based regulatory approach but supports a risk-informed approach to regulation. This approach represents a philosophy whereby risk insights are considered, together with other factors in the regulatory process, to better focus licensee and regulatory attention on design and operational issues commensurate with their importance to public health and safety. The staff does not typically use an explicit consideration of the numerical probability that a scenario would occur (i.e., number 2 of the risk triplet) to determine compliance with the LTR. This is a departure from a purely risk-based approach.

The typical deterministic approach to regulatory decision-making establishes requirements for engineering margin and for quality assurance (QA) in design, manufacture, and construction. In addition, it assumes that adverse conditions can exist and establishes a specific set of design-basis events (i.e., What can go wrong?). The deterministic approach involves implied, but unquantified, elements of probability in the selection of the specific design-basis events to be analyzed. Then, it requires that the design include safety systems capable of preventing and/or mitigating the consequences (i.e., What are the consequences?) of those design-basis events to protect public health and safety. Thus, a deterministic analysis explicitly addresses only two questions of the risk triplet.

The risk-informed approach has enhanced the deterministic approach by (1) allowing explicit consideration of a broader set of potential challenges to safety, (2) providing a logical means for prioritizing these challenges based on risk significance, operating experience, and/or engineering judgment, (3) facilitating consideration of a broader set of resources to defend against these challenges, (4) explicitly identifying and quantifying sources of uncertainty in the analysis (although such analyses do not necessarily reflect all important sources of uncertainty), and (5) leading to better decision-making by providing a means to test the sensitivity of the results to key assumptions.

Where appropriate, a risk-informed regulatory approach can also be used to reduce unnecessary conservatism in purely deterministic approaches or can be used to identify areas with insufficient conservatism in deterministic analyses and provide the bases for additional requirements or regulatory actions. Risk-informed approaches lie between the risk-based and purely deterministic approaches (SRM-SECY-98-144).

The NRC's risk-informed regulatory approach to the decommissioning of nuclear facilities is intended to focus the attention and resources of both the licensee and the NRC on the more risk-significant aspects of the decommissioning process and on the elements of the facility and the site that will most affect risk to members of the public following decommissioning. While a licensee must comply with all Commission regulations, a licensee whose site (or aspects of a site) have higher risk significance may need to provide a more rigorous demonstration to support compliance. Furthermore, the NRC staff generally will apply more scrutiny to reviews of such sites or situations with higher risk significance. This should result in a more effective and efficient regulatory process. The risk-informed regulatory approach to decommissioning is reflected in this volume, as shown by the following examples:

- The NRC has developed and is applying the concept of "decommissioning groups" based on (1) the nature and the extent of the radioactive material present at a site and (2) the complexity of the decommissioning process. The groups are generally related to the potential risks associated with the site, in that the less complex sites with limited

distribution of radioactive material may pose lower risks (i.e., manageable risks) to individuals and populations during and following decommissioning (see Section 1.2).

- The NRC's framework for decommissioning regulatory decision-making reflects the iterative nature of the compliance demonstration process. The iterative approach to decision-making for license termination provides a process for screening sites and for directing additional data collection effort toward demonstrating compliance. The framework is designed such that the level of complexity and rigor of analysis conducted for a given site should be commensurate with the level of risk posed by the site (see Section 1.4).

- This volume provides two different approaches for demonstrating compliance with the dose-based decommissioning criteria, using either a dose modeling approach or a DCGL approach. The dose modeling approach uses measurements of the actual residual radioactivity at a site after cleanup to more realistically assess the potential dose, and therefore the risk, associated with a decommissioned site. The DCGL approach allows a licensee to calculate, *a priori*, a concentration limit (DCGL) for each radionuclide based on the dose criteria of the LTR and to then demonstrate that the residual radionuclide concentrations are below the DCGLs (see Section 2.5).

- This volume allows either a screening approach or a site-specific approach to demonstrate compliance. The screening approach allows sites that pose lower potential risks to demonstrate compliance through simpler, yet conservative, screening analysis by adopting screening DCGLs developed by the NRC (see Sections 2.6 and 5.1 and Appendix H).

- The NRC staff recommends using the data quality objectives (DQOs) process for establishing criteria for data quality and developing survey designs. The process uses a graded approach to data quality requirements, based on the type of survey being designed and the risk of making a decision error based on the data collected. This process aligns the resources expended to collect and analyze data with the risk-significance of the data (see Section 3.2).

- The NRC provides for an approach to dose assessment that accounts for the site-specific risk significance of radionuclides and exposure pathways. The NRC staff allows a licensee to identify radionuclides and exposure pathways that may be considered "insignificant," based on their contribution to risk, and remove them from further consideration (see Section 3.3). The NRC endorses the MARSSIM approach to FSS design and execution, which results in a site-specific FSS design that is commensurate with potential risks associated with a site, in terms of the likelihood of exceeding the DCGLs at the site (see Section 4.4).

- The NRC staff supports a risk-informed approach to site-specific dose modeling for compliance demonstration in several ways: (1) allowing for the site-specific selection of risk-significant exposure scenarios, exposure pathways, and critical groups, (2) expecting selection of conceptual models, numerical models, and computer codes that incorporate the more risk-significant elements of a site, (3) expecting site-specific data for the more risk-significant input parameters, and allowing for more generic data for less risk-significant parameters, and (4) encouraging the use of probabilistic techniques to evaluate and quantify the magnitude and effect of uncertainties in the risk

assessment, and the sensitivity of the calculated risks to individual parameters and modeling assumptions (see Appendix I and Q).

- The NRC allows for early partial release of a portion of a site before completion of decommissioning for the entire site, based on the risks associated with the early partial site release (PSR) (see Appendix K).

- The NRC staff supports a risk-informed graded approach for engineered barriers, and this guidance includes an example of how the risk-informed approach is applied to designing erosion protection barriers (see Appendix P). In addition, the staff supports a risk-informed graded approach for selecting institutional controls and for long-term monitoring and maintenance at restricted use sites, which allows licensees to tailor the type of institutional controls and the specific restrictions on future site use based on a risk framework and insights from dose assessments (see Section 17.7 and Appendix M of Volume 1).

## 2.2 Flexibility in Submissions

The NRC staff expects that certain information will be included in licensees' DPs, including the FSS design (if an FSS will be performed) and a description of the development of DCGLs or the dose assessment, as applicable. Volume 1 of this NUREG provides additional details on the expected content in these submittals.

Some information is required by regulations (e.g., 10 CFR 30.36(g)(4)) and must be provided in the DP; the DP must include all of the following:

- the conditions of the site, building, or area, sufficient to evaluate the acceptability of the plan

- the planned decommissioning activities

- the methods used to ensure protection of workers and the environment against radiation hazards during decommissioning

- the planned final radiation survey

- an updated cost estimate for decommissioning, comparison with decommissioning funds, and a plan for ensuring the availability of adequate funds to complete decommissioning

In addition, DCGLs are usually submitted in the DP. Therefore, the typical approach is for the licensee to obtain all the detailed information needed and to submit the information in the DP. Using the DP checklist (Appendix D of Volume 1 of this NUREG report) as a guide, licensees should coordinate with the NRC staff to determine what information should be included in the DP. For example, for a MARSSIM FSS, the licensee may perform sufficient characterization surveys to determine the appropriate number of samples to obtain for each survey unit. In this case, the NRC staff could approve both the survey design and the DP, and the FSS report (FSSR) may focus primarily on the results of the FSS.

In some cases, not all of the desired information will be available during the DP preparation. For an FSS, the MARSSIM approach requires that certain information needed for the final

radiological survey be developed as part of the remedial activities at the site; therefore, this information may not be available for the DP. Similarly, some aspects of the DCGL development or dose assessment may not be available before remediation and final surveys are complete.

When some important information is not available at the time of the DP submission, licensees may either (1) make assumptions about the information or (2) commit to following a specific methodology to obtain the information. In the first case, assumptions will be considered by the NRC staff to be commitments to ensure and subsequently demonstrate that the assumption is true. The licensee would then submit the information at the completion of remediation, at the completion of FSS design, with the FSSR, or at some other appropriate time. For example, a facility uses the ratio of concentrations of thorium (Th)-232 to uranium (U)-238, along with measured concentrations of Th-232, in estimating the concentration of U-238. The licensee may have preliminary information about the ratio of concentrations and, if it is reasonable, may assume that that ratio would be valid for the conditions at the time of the FSS. The NRC staff could accept the use of the assumed value for the ratio. The licensee would demonstrate, at a later stage that the assumed value was valid, perhaps based on measurements made during the FSS.

In the second case, a licensee's DP commits to following a specific methodology to obtain the information. For example, a facility may not have sufficient information at the time of the DP submission to determine the number of samples to be taken from each survey unit for a MARSSIM FSS. In this case, the licensee may commit to the procedure recommended in MARSSIM for determining the number of samples in a survey unit and document it in the DP.

The licensee then may determine the number of samples for each survey unit as information is obtained. An FSSR could describe the FSS design, including the number of samples, which the NRC staff would evaluate, along with the FSS results.

> Depending on the circumstances and the type of information that is not specifically included in the DP, the NRC staff may consider requiring license conditions to formalize the licensee's commitments. This can be accomplished by a specific license condition or by reference to the approved DP (i.e., in the "tie-down" condition). Licensees should contact the NRC staff for the details on implementing these types of licensee commitments.

The licensee could take a similar approach to information needed to complete a dose assessment. One example is a facility for which the fraction of building-surface residual radioactivity that is removable has been determined during scoping surveys, but the licensee does not know whether remediation activities will change the fraction. In this case, the licensee might assume, for its dose assessment, that the measured fraction will remain unchanged. The NRC staff expects the licensee (1) to make measurements or calculations to demonstrate that the removable fraction was representative of the conditions when remediation is complete and (2) to demonstrate that the dose assessment is representative.

The NRC staff normally would not review DPs or FSSs that use assumptions in lieu of specific information that reasonably could be obtained before submission. In general, the NRC staff expects that assumptions used in developing DPs submitted for review would be limited to those parameters that could change as a result of the remediation or the FSS process itself or

to those parameters for which information cannot reasonably be obtained at the time of DP submission. The NRC staff should consider other assumptions on a case-by-case basis.

**Cautions on Making Assumptions or Committing to a Methodology**

If a licensee finds it reasonable to use the flexible approaches discussed here (e.g., making assumptions to be verified later or committing to a methodology to be performed later), the NRC cautions that (1) it may require a more detailed demonstration of compliance and (2) there may be a greater chance that the staff would not approve the facility's release. This is because, NRC staff would be reviewing some of the overall compliance strategy at the end of the decommissioning process rather than reviewing complete and detailed information earlier as part of the DP approval process. In addition, the licensee may be required to resolve the assumptions and commitments to meet license conditions. The licensee should contact the NRC staff for details on the use of these flexible approaches.

## 2.3 Use of Characterization Data for Final Status Surveys

Although the FSS is generally discussed as if it were an activity performed during a single stage of the radiation survey and site investigation (RSSI) process (see Chapter 4 and Table 4.1 for more about the RSSI process), this does not have to be the case. There is no requirement that an FSS be performed at the end of the decommissioning process. Data from other surveys conducted during the RSSI process—such as scoping, characterization, and remedial action support surveys—can provide valuable information for an FSS, provided the data are of sufficient quality.

In some cases, the data obtained from these other surveys may be sufficient to serve as an FSS. Licensees may plan the different phases of the RSSI to obtain data of sufficient quality to serve as or to supplement the FSS. The DQO process may be applied to all phases of the RSSI, with DQOs being as robust as those typically developed for the FSS. This approach may result in more costly characterization or remedial action support surveys (to support the more stringent DQOs), which may be balanced against the elimination of a separate FSS.

## 2.4 Choice of Null Hypothesis for Final Status Survey Statistical Analysis

The default assumption used in the MARSSIM approach to FSSs and followed by the NRC staff is that the survey unit is considered contaminated above the limit, unless survey data show otherwise. Thus, the null hypothesis used for the MARSSIM FSS statistical tests is that the concentrations of residual radioactivity exceed the DCGLs. This assumption and null hypothesis are considered Scenario A. In most all cases, the NRC staff will consider Scenario A to be the appropriate choice. In some limited cases, a different assumption and null hypothesis, Scenario B, may be appropriate. Scenario B is typically used when the DCGL is within the range of background variability making it difficult to distinguish between residual radioactivity and background. This section (and Appendix A and G, Sections A.4.4 and G.6) provide guidance on Scenario B. NUREG-1505, Revision 1, "A Proposed Nonparametric Statistical Methodology for the Design and Analysis of Final Status Decommissioning Surveys: Interim Draft Report for Comment and Use," issued June 1998, forms the basis for most of the guidance on Scenario B and can be referred to for additional details. Table 2.1 summarizes the differences between Scenarios A and B.

**Table 2.1 Comparison of FSS Statistical Test Scenarios**

| Characteristic | Scenario A | Scenario B |
|---|---|---|
| Assumption for statistical test | The survey unit is assumed to fail unless the data show it can be released.[a] | The survey unit is assumed to pass unless the data show that further remediation is necessary. |
| Null hypothesis | The concentration of residual radioactivity in the survey unit exceeds the DCGLs. | The concentration of residual radioactivity in the survey unit is indistinguishable from background. |
| Scenario emphasis | Compliance with a dose limit. | Indistinguishable from background. |
| What is needed to reject the null hypothesis? | The concentration of residual radioactivity in the survey unit is less than the DCGL. | The concentration of residual radioactivity in the survey unit is distinguishable from background. |
| Rejecting the null hypothesis means | The survey unit passes. | The survey unit fails. |
| Increasing the number of measurements in a survey unit | Increases the probability that an adequately remediated survey unit will pass. | Increases the probability that an inadequately remediated survey unit will fail. |
| When should the scenario be used? | Should be used in most cases (i.e., default) when the DCGL is fairly large compared to measurement variability. | Should be used in special cases (i.e., exception) when the DCGL is small compared to measurement and/or background variability. |
| [a] For both Scenarios A and B, "passing" the FSS means a conclusion that the survey unit may be released, and "failing" means a conclusion that the survey unit may not be released. | | |

Deciding which scenario to use and the process to make that decision are difficult issues. In most cases, when the area-wide DCGL ($DCGL_W$) for the survey unit is large compared to the measurement variability, Scenario A should be chosen. This is because even residual radioactivity below the $DCGL_W$ should be measurable. In some cases, however, it may be more appropriate to demonstrate indistinguishability from background. When the $DCGL_W$ is small compared to measurement and/or background variability, Scenario B may be appropriate. This is because residual radioactivity below the $DCGL_W$ may be difficult to measure. Background variability may be considered high when differences in estimated mean concentrations measured in potential reference areas are comparable to screening level DCGLs. Appendix G, Section G.6 of this volume provides an example of the use of Scenario B to demonstrate indistinguishability from background when the residual radioactivity consists of radionuclides that appear in background, and the variability of the background is relatively high (see also NUREG-1505 for additional details).

As mentioned above, the NRC staff's default assumption is that the use of Scenario A is appropriate. The use of Scenario B is expected only for a small number of facilities, and the

considerations for any given facility are expected to be site-specific. Therefore, the staff recommends that licensees contact the NRC early in the FSS design process to discuss considerations for their situation.

**Cautions on the Use of Scenario B for FSS Statistical Tests**

- Scenario B should typically only be used when residual radioactivity in the survey unit is within the range of background variability, making it difficult to distinguish between the residual radioactivity and background.
- Case-by-case evaluation is required.
- Licensees considering the use of Scenario B for compliance with 10 CFR Part 20, Subpart E, are strongly encouraged to contact the NRC staff early in the planning process.
- Additional information about when it is appropriate to use and potential implementation methods for Scenario B can be found in NUREG-1505 and Appendix G of this document.

## 2.5 Demonstrating Compliance Using Dose Assessment Methods versus Derived Concentration Guideline Levels and Final Status Surveys

There is flexibility in the general approach to demonstrating compliance with the dose criteria of 10 CFR Part 20, Subpart E. Two major approaches are available, the dose modeling approach and the DCGL and FSS approach. The dose modeling approach involves characterizing the site post-remediation, if remediation is necessary, and performing a dose assessment. The DCGL and FSS approach requires the development of DCGLs and performing a FSS to demonstrate that the DCGLs have been met. Because the second option is more common, emphasis is placed on the use of DCGLs and FSSs as the compliance method in this NUREG. However, the two approaches are not mutually exclusive; a hybrid of both approaches can be used. In fact, the NRC recommends licensees assess the final estimated dose using data from the FSS even if the DCGL and FSS approach is used. Table 2.2 shows some advantages and disadvantages of the two approaches.

**Table 2.2 Comparison of Dose Modeling to DCGL and FSS Approaches to Compliance**

| Approach | Advantages | Disadvantages |
|---|---|---|
| **Dose Modeling** | • more realistic<br>• accounts for time of peak dose for mixes of radionuclides<br>• can use additional data collected during decommissioning for site-specific analyses<br>• can guide remediation activities and data collection | • may still need preliminary cleanup goals or DCGLs to design surveys or guide remediation<br>• greater chance of additional iterations of remediation and/or site characterization |
| **DCGLs and FSSs** | • could be simpler to implement<br>• lower likelihood of not showing compliance with dose criterion after remediation | • by using sum of fractions, provides level of conservatism for radionuclide mix, if peak dose from individual radionuclides occurs at different times<br>• additional modeling data (i.e., to modify DCGLs) collected during decommissioning cannot be used without license amendment |

### 2.5.1 Dose Modeling Approach

Calculating the final dose is the most direct approach to show compliance with the dose criteria in Subpart E. Direct calculation of the total dose—from all radionuclides in a code that correctly accounts for the time of the peak dose for each radionuclide—is a more realistic measure of the potential dose from the site. Another advantage of this approach is that a licensee can use dose modeling information during the decommissioning process to guide additional site characterization, remediation, or other decommissioning options. Additional site characterization could be performed to reduce the level of conservatism in the dose model, parameters, or exposure scenario.

An advantage for sites that comply with the Subpart E criteria without any cleanup is that it may be unnecessary to create any DCGLs; however, the quality of the licensee's site characterization data should be sufficient for use as an FSS.

A disadvantage of the dose modeling approach is that changes in the dose modeling, between the approval of the DP and the request for license termination, would result in requiring the NRC staff to review the new information before granting approval of license termination. This additional review step could result in further justification, modeling, remediation activities, or site characterization before approval is granted. This additional review step is similar to what can occur for a site that needs no remediation but uses site-specific dose modeling to show compliance as part of the DP.

Another disadvantage of using the dose modeling approach is that cleanup goals or final concentrations may need to be estimated (1) to provide assurance that the approach will result in compliance and (2) to design quality surveys, guide remediation activities, and perform additional site characterization.

### 2.5.2 DCGL and FSS Approach

For many sites, especially those that need remediation, the DCGL and FSS approach is a simpler system to show compliance with Subpart E. The DCGL and FSS approach is the one most commonly used for compliance with the LTR and is the one MARSSIM recommends. In the DCGL and FSS approach, the licensee commits to a single concentration value for each radionuclide (i.e., $DCGL_W$) that results in a dose equal to the dose criterion. The $DCGL_W$ derivation can use either generic screening criteria or site-specific analysis. The licensee then uses FSSs to demonstrate that the DCGLs have been met. For sites with multiple radionuclides or sources, a sum of fractions approach is typically used to ensure that the dose from all radionuclides and all sources complies with the Subpart E criteria (see Section 2.7).[1] The disadvantages of this approach include the following:

- The sum of fractions approach (Section 2.7) has an underlying assumption that the peak dose for every radionuclide occurs at the same time. This can result in an additional level of conservatism, depending on the mix of radionuclides.

- Any changes in the DCGLs (e.g., because of new site information) may require a license amendment and NRC staff review.

## 2.6 Merits of Screening versus Site-Specific Dose Assessments

The advantage of screening level analyses is that they require minimal justification, characterization, and NRC staff review. The disadvantages are that (i) only residual radioactivity associated with buildings *surfaces,* and *surface* soils are considered (may not be appropriate for subsurface residual radioactivity), and (ii) in most cases, screening values are expected to be more restrictive than DCGLs derived using site-specific dose modeling. While site-specific analyses allow more flexibility in estimating the risk for a particular site compared to screening-level analyses, site-specific analyses require site-specific information, and therefore, resources must be spent on obtaining data or providing support for especially risk-significant parameter values. Changes in pathways, exposure scenarios, and conceptual models may also require supporting information. Table 2.3 provides a brief summary of the attributes and merits of both approaches.

The models, exposure scenarios, and parameters used in screening are intended to be conservative, because the lack of information about a site warrants the use of conservative models and default conditions to ensure that the derived dose is not underestimated. The screening analysis is intended to overestimate the dose, to ensure that, for 90 percent of the screening cases, the derived dose is not underestimated. In performing a screening analysis, the NRC staff should recognize that the 90th percentile of the dose distribution is used for

[1] Licensees provide their strategies and methods for compliance with LTR criteria in their DP or LTP, which typically include DCGLs and a sum of fractions approach. Once the DP or LTP is approved by NRC, it becomes part of the license. There have been cases in which the NRC has approved plans where the licensee has asked for specific criteria that allow certain narrow types of changes without NRC approval (e.g., allowance of higher clean-up levels or DCGLs under specific conditions) and that the staff found to be acceptable. Otherwise, any change to the approved DP or LTP would require NRC approval via a license amendment.

calculating compliance, whereas in the site-specific analysis, the “peak of the mean” dose over time (e.g., 1,000 years) may be used. Deterministic analyses may also be used with sufficient support for those parameters that have a significant impact on dose as identified through sensitivity analysis. An analysis is considered to be site-specific when default parameters in the DandD code used for deriving screening values are changed, source term conditions are modified, or different models or codes are used.

## 2.7 Sum of Fractions

The sum of fractions is a simple, yet flexible, approach to deal with multiple radionuclides or sources. A source is any discrete material or medium that contains residual radioactivity. For example, a site with residual radioactivity in surface soil, groundwater, and buried piping has at least three sources. The $DCGL_W$ is equivalent to the concentration of a single radionuclide from a single source that would provide 0.25 millisieverts per year (mSv/y) (25 millirem (mrem)/y) total effective dose equivalent (TEDE). The dose from each radionuclide and source should be calculated and then added together. If a licensee only complied with the $DCGL_W$ for each radionuclide in each source, the resulting total dose could be as high as 0.25 mSv/y (25 mrem/y) multiplied by the number of radionuclides multiplied by the number of sources. Unless there was only one source and one radionuclide, the resulting dose would not meet the limits detailed in Subpart E. The dose from all the radionuclides and sources must be equal to or less than the appropriate dose limit in Subpart E.

One simple way to calculate the dose from one radionuclide from one source is to calculate the relative ratio of the residual radioactivity concentration over the $DCGL_W$. Then, the ratio is multiplied by 0.25 mSv/y (25 mrem/y). In fact, for multiple sources or radionuclides, the ratios can be added together, and the sum multiplied by the dose limit. Therefore, the sum of the ratios for all the radionuclides and sources may not exceed “1” (i.e., unity). For example, if radionuclides *A* and *B* are present at respective concentrations of *Conc A* and *Conc B*, and if the respective applicable DCGLs are *Limit A* and *Limit B*, then the concentration needs to be limited so that the following relationship exists to meet Subpart E:

$$\frac{Conc\ A}{Limit\ A} + \frac{Conc\ B}{Limit\ B} \leq 1 \qquad (2\text{-}1)$$

Similarly, for multiple sources, the sum of the ratios resulting from the sum of the radionuclide contributions may not exceed unity. For example, if the site had a second source, also with radionuclides *A* and *B*, but in concentrations of *Conc* $A_0$ and *Conc* $B_0$, and DCGLs of *Limit* $A_0$ and *Limit* $B_0$, the following condition would need to be satisfied to meet Subpart E:

$$\frac{Conc\ A}{Limit\ A} + \frac{Conc\ B}{Limit\ B} + \frac{Conc\ A_0}{Limit\ A_0} + \frac{Conc\ B_0}{Limit\ B_0} \leq 1 \qquad (2\text{-}2)$$

**Table 2.3 Attributes of Screening and Site-Specific Analysis**

| Attribute | Screening | Site-Specific |
|---|---|---|
| Models/Codes | DandD Version 2 or later versions | Any model/code compatible with site conditions and approved by the NRC staff. |
| Scope of Application | Only for sites that meet the requirements for screening analysis. | Any site |
| Parameters | DandD default parameters | Site-specific (physical parameters)[2] |
| Exposure Scenarios/Pathways | DandD default exposure scenarios/pathways | Exposure scenarios/pathways may be modified, based on site conditions. |
| Dose Metric and Consideration of Uncertainty | The dose at the 90th percentile of the peak dose distribution within 1,000 years | Peak annual dose from a deterministic analysis or "peak of the mean" annual dose from a probabilistic analysis[3]. In both cases, adequate support for risk-significant[4] parameters and distributions is needed. Compliance is evaluated over a 1,000-year period. |
| The deterministic parameter set described in NUREG/CR-5512, Volume 1, and implemented in DandD Version 1 have been superseded by the parameter set described in NUREG/CR-5512, Volume 3, and implemented in DandD Version 2. DandD Version 1 did not support probabilistic analyses and used a default deterministic input parameter set. | | |

In the general form, the relationship of the ratios, commonly known as the "sum of the fractions" or the "unity rule," would be for $M$ sources ($s$) and $N$ radionuclides ($r$):

2 With respect to behavioral and metabolic parameters, default values used in DandD Version 2, and listed in NUREG/CR-5512, Volume 3, can be used to describe the average member of the critical group with minimal justification (e.g., when the site-specific exposure scenario is consistent with the screening scenario).

3 The "peak of the mean" should be used with caution if it is significantly different then the "mean of the peaks" and there is evidence of risk dilution as described in Appendix I and Q.

4 Risk-significant parameters are identified through sensitivity analysis. Appendix I contains additional details.

$$\sum_{s=1}^{M}\sum_{r=1}^{N}\frac{Conc_{s,r}}{Limit_{s,r}} \leq 1 \qquad (2\text{-}3)$$

where $Conc_{s,r}$ = the concentration of radionuclide *r* in source *s*, and
$Limit_{s,r}$ = the $DCGL_W$ value for radionuclide *r* in source *s*.

For sites with a number of radionuclides and sources, it may be easier to partition the acceptable fraction between various sources or radionuclides. For example, a licensee could commit to keeping the ratio from the groundwater to less than 25 percent of the dose limit.

One major, implicit assumption in using the sum-of-fractions approach is that peak doses for each radionuclide and source occur simultaneously. Because radionuclides can be transported through the environment at different rates, and a particular radionuclide may be dominated by a different pathway compared to another radionuclide, there are many radionuclides and contaminated media for which peak doses do not occur simultaneously. For example, radionuclides that result in predominantly external dose and are short-lived, such as cobalt (Co)-60, usually have a peak dose right after license termination. For radionuclides that result in peak dose through irrigation or drinking groundwater, the peak dose may not occur until years after license termination. When peak doses are from different radionuclides or sources occur at different times, the sum-of-fractions approach tends to overestimate that dose. In some situations, the overestimate may be significant and affect the compliance demonstration. The licensee could directly calculate the combined dose using final concentrations from the FSS to more accurately estimate the risk from the site (see Section 2.5).

## 2.8 Flexibility for Use of Institutional Controls and Engineered Barriers at Restricted Use Sites

The new guidance developed for restricted use sites includes risk-informed and performance-based approaches to institutional controls, engineered barriers, monitoring, and maintenance. These approaches not only enhance the attention to safety by being risk informed but also provide flexibility to licensees planning restricted use for a site. The approaches described allow licensees to select the most effective and efficient methods for restricting site use, designing engineered barriers to mitigate disruptive processes important to compliance, and planning monitoring and maintenance activities that are tailored to the specific site and indicators of potential disruptive processes and engineered barrier performance. Section 3.5, and Appendix P of this volume and Section 17.7 and Appendix M to Volume 1 describe these approaches.

# 3 CROSS-CUTTING ISSUES

This chapter provides guidance on several cross-cutting issues that relate to multiple aspects of surveys, characterization, and dose modeling. The issues addressed in this chapter include the following:

- transparency and traceability of compliance demonstrations
- DQO process
- insignificant radionuclides and exposure pathways
- considerations for other constraints on allowable levels of residual radioactivity
- the use of engineered barriers
- integration of radiological surveys and dose modeling for surface and subsurface soils

**Use of the Guidance in this NUREG Report**

- The suggestions in this NUREG report are only guidance, not requirements.
- Other methods for demonstrating compliance are acceptable.
- As noted in Section 5.3 of Volume 1 of this NUREG report, licensees are encouraged to have early discussions with the NRC staff in developing DPs. This is especially important when NRC guidance is limited on a specific topic. Early discussions can save licensees from following an approach that the NRC staff may find unacceptable and can clarify this guidance and identify areas where modification may be helpful for the staff's review.
- This volume refers to a number of other documents for guidance. In some cases, this volume states that the NRC staff has approved the referenced guidance. In other cases, the documents are only referenced as potentially relevant information. In these latter cases, the licensee should contact the NRC staff to determine the specific applicability to a facility, as appropriate.

## 3.1 Transparency and Traceability of Compliance Demonstrations

Licensees submit information to justify their conclusions about compliance with 10 CFR Part 20, Subpart E. Because of insufficient justification, the NRC staff has found a number of licensee submittals to be inadequate for NRC to conclude it has reasonable assurance that the license termination rule criteria can be met. This section describes some considerations for improving the thoroughness of licensee submittals. Transparency refers to arguments or calculations with descriptions sufficient for an independent reviewer to replicate. Traceability refers to the sources of information being relatable to the original source. The NRC staff encourages

licensees to submit compliance demonstrations that are transparent and traceable. This should result in more efficient and effective staff reviews.

To help ensure transparency and traceability, licensees should include the following in their justification:

- describe the sources of data
- provide only summary data, if appropriate.
- to the extent that summary data are provided, include references to detailed data and make them available to the NRC staff for review if requested (e.g., on an inspection)
- clearly describe the data, including units used, in tables and other presentations
- state the assumptions and ensure that the difference between assumptions and justified data or parameters is clear
- provide justifications for parameters or arguments, especially when employing nonstandard arguments or nondefault parameters
- describe uncertainties in data and parameters

## 3.2 Data Quality Objectives Process

Compliance demonstration is the process that leads to a decision as to whether or not a survey unit meets the release criteria. For most sites, this decision is supported by statistical tests based on the results of one or more surveys. In most cases, the initial assumption used by the NRC staff is that each survey unit is contaminated above the release criteria until proven otherwise (Scenario A). The surveys are designed to provide the information needed to reject this initial assumption. The NRC staff recommends using the Data Life Cycle as a framework for the planning, implementation, assessment, and decision-making phases of final surveys. Section 2.3 of MARSSIM discusses the major activities associated with each phase of the Data Life Cycle.

One aspect of the planning phase of the Data Life Cycle is the DQO process, which is a series of planning steps for establishing criteria for data quality and developing survey designs. The DQO process consists of seven steps:

(1) statement of the problem

(2) identification of the decision

(3) identification of inputs to the decision

(4) definition of the study boundaries

(5) development of a decision rule

(6) specification of limits on decision errors

(7) optimization of the design for obtaining data

The output from each step influences steps later in the process. Even though the DQO process is depicted as a linear sequence of steps, it is iterative in practice; the outputs of one step may lead to reconsideration of prior steps.

The DQO process uses a graded approach to data quality requirements, defined according to (1) the type of survey being designed and (2) the risk of making a decision error based on the data collected. This approach provides a more effective survey design, combined with a basis for judging the usability of the data collected. Thus, the DQO process is a flexible planning tool that licensees can use more or less intensively as the situation requires.

DQOs are qualitative and quantitative statements that satisfy all of the following by doing the following:

- clarifying the study objective
- defining the most appropriate type of data to collect
- determining the most appropriate conditions for collecting the data
- specifying limits on decision errors that will be used as the basis for establishing the quantity and quality of data needed to support the decision

Although the DQO process is generally used for surveys and the steps of an RSSI, the general concepts may also be applied to dose assessments. Chapter 5 contains additional information on data requirements related to dose modeling and DCGL development. Data of sufficient quality should be collected to ensure the technical defensibility of the modeling results focusing on risk-significant parameters. The DQO process should be used to guide data collection and analysis. Licensees are encouraged to apply the general concepts of the DQO process to all applicable parts of their compliance demonstration. The use of the DQO process can help ensure that the type, quantity, and quality of data and calculations used in decision-making will be appropriate for the intended application. Additional guidance on the use of the DQO process is in MARSSIM Section 2.3 and Appendix D and in EPA/600/R-96/055, “Guidance for the Data Quality Objectives Process,” issued August 2000.

Experience has shown that it is helpful for the licensee to identify all appropriate DQOs in planning and designing the final status survey plan (FSSP). The process of identifying the applicable DQOs ensures that the survey plan requirements, survey results, and data evaluation are of sufficient quality, quantity, and robustness to support the decision on whether the cleanup criteria have been met.

In purpose and scope, the DQO process can include a flexible approach for planning and conducting surveys and for assessing whether survey results support the conclusion that release criteria have been met. The DQO process can be an iterative process that continually reviews and integrates, as needed, new information in decision-making and the design of the final survey plan. Finally, the selection and optimization of DQOs should facilitate the later evaluation of survey results and decision-making processes during the data quality assessment (DQA) phase. Licensees have had difficulty developing DQOs, especially during the optimization step, and have not taken full advantage of the DQO process. Experience has shown that licensees often rigidly structure the process by relying too much on characterization data and are not readily open to the possibility of incorporating new information as it becomes available. This rigid approach makes implementing any changes difficult and is an inefficient

use of resources that imposes time delays (e.g., the additional time required to determine how to implement any changes).

## 3.3 Insignificant Radionuclides and Exposure Pathways

**Licensees should note that they are required to comply with the applicable dose criteria; nothing in this discussion should be interpreted to allow licensees to exceed the criteria.**

This section provides guidance on conditions under which radionuclides or exposure pathways may be considered insignificant and may be eliminated from further consideration. The dose criteria in 10 CFR Part 20, Subpart E, apply to the total dose from residual radioactivity. Thus, demonstrations of compliance should generally address the dose from all radionuclides and all exposure pathways. However, the NRC staff has determined that it is reasonable to eliminate radionuclides or pathways that are insignificant contributors to dose from further detailed consideration, although their dose contributions must be considered in demonstrating compliance with the radiological criteria for license termination.

The NRC staff considers radionuclides and exposure pathways that contribute no more than 10 percent of the dose criteria, considering uncertainty[1], to be insignificant contributors to dose. Because the dose criteria are performance criteria, this 10-percent limit is an aggregate limitation only. That is, the sum of the dose contributions from all radionuclides and pathways considered insignificant should be no more than 10 percent of the dose criteria (e.g., no more than 0.025 mSv/y (2.5 mrem/y) for the unrestricted release limit of 0.25 mSv/y (25 mrem/y)). No limitation on either single radionuclides or individual pathways is necessary. In cases of restricted release, where two dose criteria apply (one for institutional controls in place and one that considers the possibility of restrictions failing), the 10-percent limitation should be met for both dose criteria. In making a determination that radionuclides or pathways are insignificant, licensees should consider both reasonably foreseeable and less likely but plausible exposure scenarios (see Section 5 for more information). Licensees should also consider the presence of elevated areas, and potential in-growth of progeny from postulated insignificant radionuclides, when determining that the radionuclides or pathways contribute no more than 10 percent of the dose criteria and are, therefore, insignificant.

Typically, licensees would use characterization data to show that certain radionuclides or pathways are insignificant before remediation or the FSS. However, if remediation is planned, licensees may also be able to show through analysis that the dose contributions of certain radionuclides following remediation will contribute no more than 10 percent of the dose criteria.

The approach discussed in the preceding paragraph (accounting for removal of radioactivity during remediation) is similar to the use of surrogate radionuclides that implicitly considers the dose contributions of hard-to-detect radionuclides (see Appendix A of this document and Section 4.3.2 of MARSSIM, Revision 1 for additional information on use of surrogate radionuclides). However, unlike surrogate radionuclides, the licensee has an additional burden of showing that the dose contributions of the radionuclides or pathways are insignificant. Characterization data can be used to define the relative activities of significant and insignificant

[1] Uncertainty in exposure scenarios is considered through evaluation of both reasonably foreseeable and less likely but plausible scenarios. Uncertainty in parameter values can be managed through conservative assumptions (e.g., selection of parameter values from parameter distributions that tend to lead to higher doses). Chapter 5 and Appendix I have more information on consideration of uncertainty.

radionuclides to determine the relative dose contributions of the radionuclides present at the site. Ratios should be conservatively selected so that they do not underestimate the potential dose contributions of the insignificant radionuclides (e.g., use of minimum detectable concentrations (MDCs) for undetected radionuclides, and use of the 95th percentile ratios of insignificant to significant radionuclides). The licensee should also show that the relative dose contributions of the insignificant radionuclides will not increase following remediation because of an increase in concentration or redistribution of residual radioactivity. In general, the NRC does not require post-remediation sampling of the insignificant radionuclides, due to their low risk significance. However, if there is a valid concern that the dose contributions of the postulated insignificant radionuclides could be significant following remediation, licensees may choose to manage this uncertainty as part of the DQO process (e.g., through post-remediation sampling of the insignificant radionuclides, similar to the approach used for surrogate radionuclides discussed in MARSSIM Section 4.3.2).

Once a licensee has demonstrated that radionuclides or exposure pathways are insignificant, then (1) the dose from the insignificant radionuclides and pathways must be accounted for in demonstrating compliance, but (2) the insignificant radionuclides and pathways may be eliminated from further detailed evaluations. For example, after sufficient site characterization, suppose a licensee shows that the dose from strontium (Sr)-90 at the facility is 0.02 mSv/y (2 mrem/y), which is less than 10 percent of the dose criterion for unrestricted use. In this case, Sr-90 can be considered insignificant and eliminated from the FSS and from detailed consideration in the dose modeling. However, the dose from Sr-90 has to be considered in demonstrating compliance with the dose criterion. In some cases, licensees may be able to show that the dose contributions of the insignificant radionuclides and pathways are much less than 10 percent of the dose criteria. However, it may be less burdensome on the licensee to show that the insignificant radionuclides and pathways contribute less than 0.025 mSv/y (2.5 mrem/y) than it is to show that the insignificant radionuclides and pathways contribute much less than 0.01 mSv/y (1 mrem/y), for example. Thus, it may be preferable to round the dose contributions of the insignificant radionuclides and pathways higher, and thereby provide a greater safety margin in meeting the dose criteria.

It is important for the licensee to document the radionuclides and pathways that it has considered insignificant and eliminated from further consideration and for the licensee to justify the decision to consider them insignificant. However, licensees and the NRC staff should be aware that remediation techniques (or other activities or processes) may result in an increased dose from the postulated insignificant radionuclides or pathways, such that the dose contributions are no longer insignificant. Thus, licensees should also demonstrate that the dose contributions of insignificant radionuclides and pathways deemed insignificant will not increase (or were not underestimated) as a result of other activities.

**Summary of Determining Insignificant Radionuclides and Exposure Pathways**

- Licensees may eliminate insignificant radionuclides and exposure pathways from further detailed consideration. However, the dose from the insignificant radionuclides and pathways must be accounted for in demonstrating compliance with the applicable dose criteria.
- Insignificant means no greater than 10 percent of applicable dose criteria.
- Ten percent is an aggregate limit; total dose contributions of all radionuclides and all exposure pathways considered insignificant should not exceed the 10 percent limitation.
- There is no additional limit on single radionuclides or pathways.
- Licensees should also address the potential for dose from postulated insignificant radionuclides or pathways to increase during remediation activities.

## 3.4 Considerations for Other Constraints on Allowable Residual Radioactivity

Situations or standards other than the dose criteria and ALARA requirements of Subpart E may constrain the final dose below 0.25 mSv/y (25 mrem/y). Two main causes for constraining the Subpart E dose limit are (1) a PSR and (2) other standards or regulations.

A PSR occurs when a licensee releases a portion of its site for unrestricted use before terminating the entire license. While the licensee should demonstrate that the residual radioactivity at the time of unrestricted release of the specific area meets the Subpart E dose limit, the residual radioactivity of the area should also be taken into account during final termination to demonstrate that the entire site has met the appropriate release criteria. Appendix K of this volume discusses dose modeling considerations for PSR. In general, the comments below are also applicable to PSRs.

Demonstrating compliance with the Subpart E dose limit does not eliminate the licensee's requirement for meeting other applicable Federal, State, or local rules and regulations. These regulations from other governmental agencies may conflict with the requirements of Subpart E, as they may allow higher or lower levels of residual radioactivity on the site or may conflict in other ways, such as limiting decommissioning options or final status. Nevertheless, the staff should review a DP for compliance only with NRC requirements, including 10 CFR Part 20, Subpart D, which incorporates, where applicable, the requirements of 40 CFR Part 190, "Environmental Radiation Protection Standards for Nuclear Power Operations." For example, in reviewing the appropriateness of proposed DCGLs or the number of samples per survey unit for an unrestricted site, the NRC staff would use the limit of 0.25 mSv/y (25 mrem/y), and not a State's limit of 0.2 mSv/y (20 mrem/y). Thus, any requests for additional information would also be based on compliance with the limit of 0.25 mSv/y (25 mrem/y). Licensees should note that use of a lower dose standard does not necessarily lead to a lower clean-up level because of differences in assessment approaches including differences in exposure scenarios, models and parameters used by different agencies setting the dose standards.

## 3.5 Use of Engineered Barriers

This section, and the details provided in Appendix P, provide guidance to licensees considering the use of engineered barriers (e.g., engineered covers, including those designed for erosion protection, stabilizing cementitious materials, and reactive walls) to demonstrate compliance with radiological criteria for license termination. Section 3.5 and Appendix P also support Section 17.7.3 of Volume 1 by giving guidance on the information expected to be submitted in a DP for the engineered barrier analysis and the technical basis for engineered barrier performance.

In the Commission's view, engineered barriers are distinct and separate from institutional controls (see *Decommissioning Criteria for the West Valley Demonstration Project (M-32) at the West Valley Site: Final Policy Statement* (67 FR 5003)). Generally, institutional controls are designed to restrict access, whereas engineered barriers are usually designed to inhibit water from contacting waste, limit releases, or mitigate doses to intruders.

Engineered barriers are passive, man-made structures or devices intended to enhance a facility's ability to meet the dose criteria in the LTR.[2] Engineered barriers are usually designed to inhibit water from contacting waste and releasing radionuclides to groundwater, thereby reducing exposure from ingestion of contaminated water. Engineered surface barriers may also slow erosion or otherwise decrease the likelihood of the waste being exposed at the surface through human or biotic activity, thereby reducing dose from inadvertent intrusion and direct exposure to the waste. In some cases, engineered barriers may also be used to passively limit access of critical groups to residual radioactivity (e.g., a durable rock barrier that decreases the likelihood of excavation or drilling into residual radioactivity). Used in the general sense, an engineered barrier could be one of a broad range of barriers with varying degrees of durability, robustness, and isolation capability.

On the other hand, institutional controls are active controls that limit access to the site, and the use of it, to minimize disturbances to engineered barriers and ensure that the exposure from the residual radioactivity does not exceed the established criteria. Institutional controls include legal mechanisms (e.g., land use restrictions) and physical controls (e.g., signs, markers, landscaping, and fences) to prevent unauthorized access to the site and minimize disturbances to engineered barriers. Institutional controls may require financial assurance to ensure adequate control and maintenance of the site, legal enforceability, and an entity with the capability, authority, and willingness to enforce the controls.

The licensee determines the functionality and robustness of barriers using the risk-informed graded approach described in Appendix P and evaluated on a site-specific basis for each license application. However, the general framework that a licensee should consider would not vary from licensee to licensee; only the depth and breadth of information supplied to demonstrate the performance of the engineered barriers may vary. Appendix P provides the general framework a licensee should consider for use of engineered barriers in the decommissioning process.

It is expected that engineered barriers will most frequently be used for restricted use sites. However, there may be infrequent cases where engineered barriers are used as one

[2] In some cases, for restricted-use sites, a licensee can propose active monitoring and maintenance of the engineered barrier; this would be considered an institutional control and could be used to enhance the assumed level of performance or longevity of the engineered barrier beyond its passive performance.

component of a decommissioning approach to achieve unrestricted use of a site. These cases should be infrequent because of the uncertainty associated with the long-term performance of engineered systems without monitoring and maintenance and because the goal should be to achieve unrestricted use without relying on engineered barriers. If an engineered barrier is used at an unrestricted use site, only the passive performance of the barrier to mitigate radiological impacts may be credited (i.e., performance of the barrier without monitoring, inspection, and maintenance) in the dose assessment to demonstrate compliance with the LTR dose criteria. The assessment of engineered barrier performance should consider reasonably foreseeable, as well as less likely but plausible, disruptive conditions from human activities and from natural events and processes. Results of reasonably foreseeable exposure scenarios should be strictly considered when demonstrating compliance with radiological criteria for license termination, while less likely but plausible exposure scenarios should also be considered to help risk-inform the decision-making process.

Proposals to use engineered barriers for unrestricted use sites should be part of an overall plan for decontaminating and decommissioning a site, as presented in a licensee's DP or LTP. The licensee should demonstrate that, at the time of license termination, the site meets LTR criteria, including the criterion to reduce residual radioactivity to ALARA levels. Therefore, licensees should consider whether the following would be consistent with ALARA requirements in the absence of engineered barriers: (1) removal and disposal of contaminated components and equipment, (2) decontamination (and demolition, if appropriate) of buildings, (3) removal and disposal of waste streams remaining on site from past operations, and (4) excavation and removal of large areas of soil contamination as waste. Therefore, while engineered barriers may be proposed to assist with meeting the LTR criteria, source removal must first be considered in demonstrating that residual radioactivity has been removed to ALARA levels. Chapter 6 and Appendix N contain additional information.

Similarly, for restricted use sites, under 10 CFR 20.1403(a) the licensee must show that further removal of residual radioactivity would result in net public or environmental harm or that leaving the residual radioactivity in place is ALARA. Licensees should also include other considerations (e.g., distance to disposal facility, efficient use of available disposal capacity at the offsite facility, unavailability of required treatment options, lack of disposal options other than leaving the contaminated materials on site, and the need to use funds to remediate nonradioactive hazards at the same site), if applicable and appropriate, in its determination of whether additional removal of residual radioactivity is reasonably achievable.[3] In their proposal to use engineered barriers, licensees should include all relevant information concerning the risks of using the proposed approach versus other remediation alternatives.

Because of the wide range of licensed decommissioning sites, the LTR and the NRC's decommissioning guidance are not prescriptive as to the criteria for, or acceptability of, site-specific engineered barriers. Therefore, the licensee has flexibility in the methods used to demonstrate compliance with the performance-based criteria of 10 CFR Part 20, Subpart E. Because of this flexibility and because engineered barrier designs are site-specific, it is very important for the licensee to clearly and completely document how it has considered site-specific conditions (e.g., site-specific resources, climate, degradation mechanisms) in its engineered barrier designs and monitoring and maintenance program.

3 "Reasonably achievable" is judged by considering the state of technology and the economics of improvements in relation to all the benefits of these improvements. See Section N.1 for additional information.

Appendix P provides the detailed framework for applying engineered barriers to achieve decommissioning at a site, and an example of a graded approach to erosion covers. A summary of existing guidance and reference information on the application of engineered barriers at decommissioning sites is also listed in Appendix P.

In summary, the following points should be considered when applying engineered barriers to achieve decommissioning at a site:

- Engineered barriers are distinct and separate from institutional controls.
- Only the passive performance of engineered barriers (i.e., no monitoring and maintenance) can be relied on at unrestricted use sites.
- Passive performance and (when institutional controls are assumed to be in place) active monitoring and maintenance of engineered barriers can be relied on at restricted use sites. Active monitoring and maintenance are considered institutional controls and may enhance the level of performance assumed for an engineered barrier beyond what is assumed for a barrier's passive performance. For most cover systems, long-term, passive performance is not well understood.
- For licensees pursuing unrestricted use of their site, residual radioactivity has been reduced to levels that are ALARA before reliance on engineered barriers to meet LTR criteria. Furthermore, for licensees pursuing restricted use of their site, a demonstration is required to show that additional removal of residual radioactivity to meet unrestricted use criteria would result in net public or environmental harm or that further reductions are not being made because levels that meet restricted use conditions are ALARA.
- Engineered barrier evaluation are reviewed on a case-by-case basis using a risk-informed approach.

## 3.6 Surveying and Considering Risk Associated with Surface and Subsurface Soils

Attempts have been made to define "surface soils," based on the capability of relatively low-cost scan instrumentation to detect residual radioactivity near the surface of buildings and in soils.[4] MARSSIM survey protocols, discussed in more detail in Chapter 4 and Appendices A–G, assume that residual radioactivity is present on the surfaces of buildings and soils. Likewise, certain dose modeling codes such as DandD, discussed in more detail in Chapter 5 and Appendices H and I, also make assumptions about the depth of residual radioactivity that could be important for assessing dose to potential receptors. For example, surface soils are important for assessing dose from certain exposure pathways such as direct radiation exposure, incidental soil ingestion, and inhalation. While surface soils could also support other important exposure pathways, such as plant growth and consumption, contaminated subsurface soils could also contribute significantly to radionuclide uptake and dose dependent on the plant type and radionuclide. Thus, if subsurface residual radioactivity is ignored, the dose could be significantly underestimated.

[4] For example, surface soil has been associated with the top 15 cm of soil that can typically be measured using scan instrumentation.

Human activities could cause subsurface soils to become surface soils from mixing or redistribution after license termination. For example, tilling soil to promote crop growth or excavating soil for home construction or well drilling could lead to the redistribution of subsurface soils to the surface (see Appendix J). Before license termination, remediation of the site and sorting of relatively contaminated and uncontaminated soils could lead to the redistribution or reuse of contaminated subsurface soils on the surface (see Appendix G). While the definition of surface soil is an important consideration when it comes to surveying and assessing the dose impacts associated with residual radioactivity remaining at a site at the time of license termination, there is no clear line of demarcation of surface and subsurface soils; the final distribution of residual radioactivity should be considered in dose modeling and the final status survey design. Potential redistribution of radioactivity following license termination must also be considered when deriving subsurface soil DCGLs or when performing dose modeling to assess the impact of subsurface residual radioactivity (Appendix J).

Despite the linkages to scan instrument capability and dose modeling discussed above, the depth of residual radioactivity in surface and subsurface soils assumed for the purpose of the FSS design should not be arbitrarily selected based on dose modeling assumptions or scan capability. Rather, the expected horizontal and vertical extent of residual radioactivity informed by characterization surveys should be considered in the FSS design and in assessing dose. Significant heterogeneity in residual radioactivity concentrations should also be considered when designing the FSS and assessing dose. Chapters 4 and Appendices A–G, provide guidance on for performing radiological surveys for surface and subsurface soils; building materials; and surface water and groundwater (Appendix F); Chapter 5, and Appendices I and J have guidance on evaluating the dose impacts associated with surface, subsurface, and heterogeneous residual radioactivity in soils and building surfaces.

If subsurface residual radioactivity is present, soil sampling will likely be necessary to supplement scan surveys to adequately characterize the full vertical extent of residual radioactivity. Depth discrete sampling of soils may also be needed if there is significant vertical heterogeneity, and representation of the vertical heterogeneity is important in assessing dose. DCGLs derived from dose modeling should be consistent with the actual vertical extent of the residual radioactivity and significant vertical heterogeneity considered. For example, it may be important to differentiate soil layers based on vertical heterogeneity and derive more than one set of DCGLs (e.g., derive surface and subsurface soil DCGLs), if vertical heterogeneity is found to be important to dose. Alternatively, dose modeling could be used to confirm that radiological criteria for license termination are met based on the final configuration of residual radioactivity in soils, as measured in FSSs. Chapter 5 and Appendix I contain additional information on the dose modeling approach. Appendix J contains information on scenarios that should be considered for buried residual radioactivity. Appendix I, Section I.2 specifically contains additional information on source term abstraction for heterogenous distributions.

# 4 FACILITY RADIATION SURVEYS

## 4.1 Radiation Survey and Site Investigation Process

As a framework for collecting the information required for demonstrating compliance identified using the DQO process (see Section 3.2 of this volume), the NRC staff recommends using a series of surveys. The RSSI process is an example of a series of surveys designed to demonstrate compliance with the decommissioning regulations of 10 CFR Part 20, Subpart E. Table 4.1 identifies the steps in the RSSI process and indicates where specific guidance on each step can be found.

**Table 4.1 Cross-References for Principle Steps in the Radiation Survey and Site Investigation Process**

| Principal Step | Applicable Guidance |
|---|---|
| Site Identification | Chapter 16, Volume 1, of this NUREG report<br>Section 2.4 of MARSSIM, Revision 1 |
| Historical Site Assessment | Section 4.1.1 of this volume<br>Section 2.4 and Chapter 3 of MARSSIM, Revision 1 |
| Scoping and Characterization Survey | Section 4.2 of this volume<br>Sections 2.4, 5.2 and 5.3 of MARSSIM, Revision 1 |
| Remedial Action Support Survey | Section 4.3 of this volume<br>Sections 2.4 and 5.4 of MARSSIM, Revision 1 |
| Final Status Survey | Section 4.4 of this volume<br>Sections 2.4 and 5.5 of MARSSIM, Revision 1 |

Note: As of the date of publication of this NUREG, MARSSIM, Revision 2, is in the process of being published. Because MARSSIM, Revision 2, has not yet been published, references to sections of MARSSIM in this volume are with respect to MARSSIM, Revision 1 (i.e., section numbers may be different in Revision 2). NRC staff plans to incorporate MARSSIM, Revision 2, into a future revision of this volume.

### 4.1.1 Historical Site Assessment

The RSSI process uses a graded approach that starts with Site Identification and the HSA and is later followed by other surveys that lead to the FSS. In most cases, the radiological status of a site will already be known based on its prior use and the presence of radioactive material although a records review could identify areas of use, disposal, or spills that have been overlooked. The HSA collects existing information describing a site's complete history from the start of site activities to the present. The necessity for detailed information and the amount of effort to conduct an HSA depend on the type of site, associated historical events, regulatory framework, and availability of documented information. The main purpose of the HSA is to determine the current status of the site or facility, but the data collected may also be used to differentiate sites that need further action from those that pose little or no threat to human health and the environment (see Section 2.3). This screening process can provide a site disposition recommendation or propose additional surveys. Because much of the data collected during

HSA activities are qualitative or are analytical data of unknown quality, many decisions on a site are the result of professional judgment.

The primary objectives of the HSA include the following:

- identify potential sources of residual radioactivity
- determine if sites pose a threat to human health and the environment
- differentiate impacted from nonimpacted areas
- provide input to scoping and characterization survey designs
- assess the likelihood of residual radioactivity migration
- identify additional potential radiation sites related to the site being investigated

The HSA typically consists of three phases: (1) preliminary investigation of the facility or site, (2) site visits or inspections, and (3) an evaluation of the site based on the information collected. The HSA should identify special survey situations that may need to be addressed, such as subsurface radioactivity; sewer systems, waste plumbing, and floor drains; ventilation ducts; and embedded piping containing residual radioactivity. Appendix G of this volume includes information on special survey situations. Section 2.4.2 and Chapter 3 of MARSSIM contain additional guidance on the HSA.

### 4.1.2 Summary of Survey Types

The NRC's regulations (10 CFR 20.1501(a)) require a licensee to make or cause to be made surveys that may be necessary for the licensee to comply with the regulations in Part 20, including the radiological criteria for license termination found in 10 CFR Part 20, Subpart E. The licensee would demonstrate compliance with this requirement by performing a FSS. The FSS will demonstrate that the licensee's site or facility, or both, can meet the radiological criteria for license termination.

Other surveys (e.g., scoping surveys, characterization surveys, and remedial action support surveys) are used to identify areas with residual radioactivity but are typically not used to demonstrate compliance with the radiological criteria for license termination.

The NRC endorses the FSS methodology described in MARSSIM. The guidance in this chapter does not replace MARSSIM, and users of this chapter should be familiar with and use MARSSIM. Thus, it is intended that licensees will use this chapter and MARSSIM as guidance for acceptable approaches or methodologies to conduct surveys supporting decommissioning and FSSs, in particular. The following text refers to specific sections of MARSSIM, Revision 1, when applicable.

The measurement methods applied in assessing radiation and radioactivity levels can vary according to the objectives of the particular survey. It is expected that different types of surveys would be conducted during the course of decommissioning work, with each having a different emphasis while at the same time, sharing common elements. The sections below summarize six survey types.

*Background Survey*—Although not specifically identified as a step in the RSSI process, this survey constitutes measurements of sites in areas surrounding the facility to establish the baseline; that is, the normal background levels of radiation and radioactivity. In some situations, historical measurements may be available from surveys performed before the construction and operation of a facility. The background survey takes on added importance if one may ultimately be comparing onsite cleanup units to offsite reference areas. Appendix A of this volume contains guidance on background surveys.

*Scoping Survey*—This survey, performed to augment the HSA, provides sufficient information (1) to determine if residual radioactivity is present that warrants further evaluation and (2) to make initial estimates of the level of effort required for remediation and to prepare a plan for a more detailed survey, such as a characterization survey. The scoping survey does not require that all radiological parameters be assessed. Sections 2.4 and 5.2 of MARSSIM and Section 4.2 of this volume contain additional guidance on the scoping survey.

*Characterization Survey*—This survey determines the type and extent of residual radioactivity on or in structures and environmental media. The survey should be sufficiently detailed to provide data for planning decommissioning actions, including remediation techniques, projected schedules, costs, waste volumes, and health and safety considerations during remediation. Section 4.2 of this volume contains additional guidance on characterization surveys.

*Remedial Action Support Survey*—This survey, which could be repetitive in nature, is conducted in what is effectively a real-time mode to guide cleanup efforts and ensure the health and safety of workers and the public. The effectiveness of the remediation efforts can be assessed as they progress. The precision and accuracy of measurements associated with this type of survey are generally not sufficient to determine the final radiological status of the site.[1] Section 4.3 of this volume contains additional guidance on remedial action support surveys.

*Final Status Survey*—This survey demonstrates that residual radiological conditions satisfy the predetermined criteria for release for unrestricted use or, where appropriate, for use with designated restrictions. It is this survey that provides data to demonstrate that all radiological parameters (e.g., total surface activity, removable surface activity, exposure rate, and radionuclide concentrations in soil and other materials) satisfy the established guidelines and conditions. Section 4.4 of this volume contains additional guidance on FSSs.

*Confirmatory Survey*—The regulator performs this survey to obtain data to substantiate the results of the licensee's FSS. The objective of this type of survey is to verify that characterization, remediation, and final status actions and documentation, conducted as part of the RSSI process, are adequate to demonstrate that the site is radiologically acceptable, relative to applicable criteria. Section 15.4.5 of Volume 1 of this NUREG report contains additional information on confirmatory surveys.

These types of surveys are performed at various stages of the decommissioning process. Early on, where known residual radioactivity exists, the simplest of measurement approaches can be used to document the need to clean up a specific building surface or parcel of land. In practice, the simpler methods would generally be applicable to the scoping and remedial action support surveys. The more complex methods, which produce data with higher precision and accuracy, will be required for background, characterization, final status, and confirmatory surveys. In

[1] In certain cases, it may be prudent to collect data of sufficient quality during the remedial action support survey to support the final status survey during remediation as discussed in Sections 4.3 and G.3.2.

general, wherever measurements are to be performed at or close to background levels, greater sensitivity in the measurement is required.

The conduct of these surveys and the methods applied have some interchangeable elements. It is possible that measurements collected in one survey can be used for another. For instance, if measurements sufficient in spatial coverage and with adequate detection limits were taken, the results of the scoping survey in an unaffected area could be used to support the FSS. The emphasis of the guidance in this volume is on the methods that can be applied to meet the requirements of the FSS, although they can be applied to other survey work as well. In performing decommissioning surveys, licensees should be cognizant of the survey methodologies and their limitations, especially where newer technologies are employed such as *in situ* gamma spectroscopy. Some information on the capabilities of this technology to detect discrete particles in soil can be found in the NRC sponsored study/report by the Oak Ridge Institute for Science and Education (ORISE) titled "Spatially-Dependent Measurements of Surface and Near-Surface Radioactive Material Using In situ Gamma Ray Spectrometry (ISGRS) For Final Status Surveys" (Chapman et al., 2006).

NRC released the final version of the Multi-Agency Radiological Laboratory Analytical Protocols manual (MARLAP) in 2004. The MARSSIM and the MARLAP manual are complementary guidance documents in support of cleanup and decommissioning activities. The MARSSIM document contains guidance on how to plan and carry out a study to demonstrate that a site meets appropriate release criteria. It describes a methodology for planning, conducting, evaluating, and documenting environmental radiation surveys conducted to demonstrate compliance with cleanup criteria. The MARLAP manual provides guidance and a framework for both project planners and laboratory personnel to ensure that radioanalytical data will meet the needs and requirements of cleanup and decommissioning activities.

The MARLAP manual recommends the use of a directed or systematic planning process. A directed planning process is an approach for setting well-defined, achievable objectives and developing a cost effective, technically sound sampling and analysis design that balances the data user's tolerance for uncertainty in the decision process with the resources available for obtaining data to support a decision. For example, the NRC and licensees have determined that side-by-side surveys (with subsequent PSRs) are more efficient than waiting for a final sitewide confirmatory survey. The NRC and licensees should plan ahead and coordinate their schedules to implement efficient side-by-side confirmatory surveys. Appendix D contains more details on MARLAP and how it can enhance radiation monitoring.

Appendix D of this volume includes information on survey data quality and reporting, Chapter 5 of MARSSIM provides survey checklists, Appendix E contains information on survey measurements, and Appendix G has information on special survey issues.

### 4.1.3 Areas of Review

The NRC staff should review the results of the radiological characterization survey to determine whether it contains sufficient information to permit planning for site remediation that will be effective and will not endanger the remediation workers, to demonstrate that it is unlikely that significant quantities of residual radioactivity have gone undetected, and to provide information that will be used to design the FSS.

The purpose of the NRC staff review is to verify that the FSS design is adequate to demonstrate compliance with the radiological criteria for license termination.

The FSS review should determine whether the results demonstrate that the site, area, or building meets the radiological criteria for license termination.

The staff should note that NRC regulations require that DPs include a description of the planned final radiological survey. Recognizing the flexible approach discussed in Section 2.2 of this volume and that the MARSSIM approach allows certain information needed to develop the final radiological survey to be obtained as part of the remedial activities at the site, a licensee or responsible party should submit information on facility radiation surveys in one of two ways, as summarized below. Section 2.2 of this volume provides additional relevant guidance.

Method 1:
The licensee or responsible party may submit the information contained in Sections 4.1–4.3 of this volume as part of the DP, along with a commitment to use the MARSSIM approach in developing the final status survey. The licensee or responsible party would then submit the information discussed in Section 4.4 at the completion of remediation or design development for the final status survey for the site. The licensee or responsible party will submit the FSSR (Section 4.5) after performing the final status survey.

Method 2:
The licensee or responsible party may submit the information contained in Sections 4.1–4.4 of this volume, along with a commitment to calculate the number of sampling points that will be used in the final status survey, in accordance with the procedure described in MARSSIM. The licensee or responsible party would then submit the FSSR (Section 4.5) after performing the final status survey. If this method is used, the licensee or responsible party should include in the FSSR the information contained in the last three bullets under "Information to Be Submitted," in Section 4.4 of this chapter.

#### *4.1.3.1 Acceptance Review*

The review should ensure that the licensee's submittal contains the information summarized under the "Areas of Review," as appropriate for the particular submittal. The NRC staff should ensure that the level of detail appears to be adequate for it to perform a detailed technical review but should not review the technical adequacy of the information, which it should determine during the detailed review.

#### *4.1.3.2 Safety Evaluation*

The material to be reviewed is both informational in nature and requires specific detailed technical analysis. The NRC staff should verify that the survey designs and results are adequate for demonstrating compliance with the radiological criteria for license termination.

### 4.1.4 Release Criteria

The NRC staff review is to verify that the licensee has provided appropriate release criteria, referred to as the DCGLs. Generally, the licensee should provide the $DCGL_W$, for the survey unit average concentrations, and the applicable $DCGL_{EMC}$ (elevated measurement comparison) for small areas of elevated concentrations, for all affected media.

#### *4.1.4.1 Acceptance Criteria*

##### *4.1.4.1.1 Regulatory Requirements*

10 CFR 20.1402, 20.1403, and 20.1404

##### *4.1.4.1.2 Regulatory Guidance*

NUREG-1575, "Multi-Agency Radiological Survey and Site Investigation Manual (MARSSIM)"

##### *4.1.4.1.3 Information to Be Submitted*

The licensee should list the DCGL(s) that will be used to design the surveys and to demonstrate compliance with the radiological criteria for release, including the following:

- a summary table or list of the $DCGL_W$ for each radionuclide and affected medium of concern
- the $DCGL_{EMC}$ (or areas factor derived from dose modeling) for each radionuclide and media of concern if Class 1 survey units are present (Appendix A.2 of this volume discusses classification of site areas)
- the appropriate $DCGL_W$ for the survey method to be used if multiple radionuclides are present

The information to be submitted is also included as part of the master DP checklist provided in this NUREG report (see Section XIV.a from Appendix D of Volume 1).

#### *4.1.4.2 Evaluation Criteria*

The NRC staff should verify that, for each radionuclide and affected media of concern, the licensee has provided a $DCGL_W$ and, if Class 1 survey units are present, a table of $DCGL_{EMC}$s (or area factors). The NRC staff should verify that the values presented are consistent with the values developed pursuant to dose modeling, as discussed in Chapter 5 of this volume. If multiple radionuclides are present, MARSSIM Sections 4.3.2, 4.3.3, and 4.3.4 describe acceptable methods to determine DCGLs appropriate for the survey technique.

## 4.2 Scoping and Characterization Surveys

### 4.2.1 Scoping Surveys

Early in the decommissioning process, the licensee identifies the potential residual radioactivity present at the site, the relative ratios of radionuclides, and the general extent of residual radioactivity—if any—both in activity levels and affected area or volume. Although the license and operational history documentation will assist to varying degrees in providing this information, it will often be necessary to supplement it with actual survey data. Therefore, the licensee shall perform a scoping survey typically consisting of limited direct measurements (exposure rates and surface activity levels) and samples (smears, soil, water, and material with induced activity) obtained (1) from site locations considered to be the most likely to contain residual activity and (2) from other site locations, including those immediately adjacent to the

radioactive materials use areas. This survey provides a preliminary assessment of site conditions, relative to guideline values. The scoping survey forms the basis for initial estimates of the level of effort required for decommissioning and for planning the characterization survey. Measurements and sampling in known areas of residual radioactivity do not need to be as comprehensive or be performed to the same sensitivity level as will be required for the characterization survey or FSS. However, when planning and conducting the scoping survey, the licensee should remember that some of the data, particularly from locations not affected by site operations, may be used as final survey results or to supplement the characterization or final survey results, or both. Similar measuring and sampling techniques as used for those categories of surveys, therefore, may be warranted.

Scoping surveys provide site-specific information based on limited measurements. The following are the primary objectives of a scoping survey:

- perform a preliminary hazard assessment
- support classification of all or part of the site as a Class 3 area
- evaluate whether the survey plan can be optimized for use in either the characterization or final stage
- perform radiological status surveys
- provide data to address the requirements of other applicable regulations
- provide input to the characterization survey design, if necessary

Scoping surveys are conducted after the HSA is completed and consist of judgment measurements based on the HSA data. If the results of the HSA indicate that an area is Class 3 and no residual radioactivity is found, the licensee may classify the area as Class 3 and conduct a Class 3 FSS. If the scoping survey locates residual radioactivity, the licensee may consider the area to be Class 1 (or Class 2) for the FSS and typically conduct a characterization survey, collecting sufficient information to identify situations that require immediate radiological attention. Licensees should be aware that requirements of other applicable regulations (e.g., non-radiological constituents) may differ from NRC requirements. Appendix F to MARSSIM contains a comparison of MARSSIM guidance to other requirements.

### 4.2.2 Characterization Surveys

After identifying the affected locations, the licensee conducts a characterization survey to more precisely define the extent and magnitude of residual radioactivity. The survey should be sufficiently detailed to provide data for planning the remediation effort, including the remediation techniques, schedules, costs, and waste volumes, as well as necessary health and safety considerations during remediation. The type of information obtained from a characterization survey is often limited to that which is necessary to differentiate a surface or area as containing or not containing residual radioactivity. A high degree of accuracy may not be required for such a decision when the data indicate levels well above the guidelines. On the other hand, when data are near the guideline values, a higher degree of accuracy is usually necessary to ensure the appropriate decision about the true radiological conditions. In addition, one category of radiological data (e.g., soil radionuclide concentration or total surface activity) may be sufficient to determine if there is residual radioactivity, and other measurements (e.g., exposure rates or

removable residual radioactivity levels) may, therefore, not need to be performed during characterization. As the scoping survey example demonstrates, the choice of survey technique should be commensurate with the intended use of the data, including considerations for possible future use of the results to supplement the FSS data.

Licensees typically submit site characterization summary information as part of their DP. If submitted site characterization information is insufficient to reasonably identify the extent and nature of residual radioactivity, the NRC may decline to accept and review the DP until such information is provided. The NRC may ask the licensee to submit site characterization plans or other site characterization information before submitting the DP, or the NRC may elect to meet with the licensee before or during site characterization work. However, licensees are not required to submit a separate site characterization or site characterization report unless required by a license condition. Rather, site characterization information is required as a component of the DP. Therefore, the NRC staff will only request site characterization and reports separate from the DP submittal when necessary to provide assurance the extent and nature of residual radioactivity are reasonably identified.

The characterization survey is generally the most comprehensive of all the survey types and generates the most data. This includes preparing a reference grid, systematic as well as judgment measurements, and surveys of different media to include surface soils and interior and exterior surfaces of buildings. Additionally, the characterization survey should identify all activated materials (typically Decommissioning Groups 4–7) and hard-to-detect radionuclides throughout the site. The decision as to which media will be surveyed is a site-specific decision addressed throughout the RSSI process (see MARSSIM).

Characterization surveys may be performed to satisfy a number of specific objectives.

Examples include the following:

- determining the nature and extent of residual radioactivity
- evaluating remediation alternatives (e.g., unrestricted use, restricted use, onsite disposal, offsite disposal)
- developing input to pathway analysis and dose or risk assessment models for determining site-specific DCGLs in becquerel/kilogram (Bq/kg), picocuries/gram (pCi/g), becquerel/square meter ($Bq/m^2$), or disintegrations per minute/100 square centimeters (dpm/100 $cm^2$), as applicable
- estimating the occupational and public health and safety impacts during decommissioning
- evaluating remediation technologies
- developing input to the FSS design
- complying with requirements of other applicable regulations

This volume does not include detailed discussions of characterization survey design for each of these objectives; the user should consult other references for specific characterization survey objectives not covered. For example, DOE's "Decommissioning Handbook," issued

March 1994, is a good reference for characterization objectives for evaluating remediation technologies or unrestricted or restricted use alternatives. Additionally, ANSI N13.59:2008, "Characterization in Support of Decommissioning Using the Data Quality Objectives Process" is a useful document which presents characterization strategies based on the DQO process. The licensee should consult other references (EPA, "Guidance for Conducting Remedial Investigations and Feasibility Studies Under CERCLA," issued October 1988; EPA, "Superfund Removal Procedures," 1988; "EPA, Federal Radiation Protection Guidance for Exposure of the General Public," dated December 23, 1994; NUREG-1501, "Background as a Residual Radioactivity Criterion for Decommissioning—Draft Report," issued August 1994) for planning decommissioning actions (e.g., remediation techniques, projected schedules, costs, and waste volumes) and health and safety considerations during remediation. Also, the specific modeling code documentation should determine the types of characterization data needed to support risk or dose modeling.

### 4.2.3 Areas of Review

The purpose of the NRC staff review is to verify that the licensee determined the radiological condition of the property well enough to permit planning for a remediation that will be effective and will not endanger the remediation workers, to demonstrate that it is unlikely that significant quantities of residual radioactivity have gone undetected, and to provide sufficient information for designing the FSS. Note that some licensees have used, or may request authorization to use, information developed during the characterization survey to support the final radiological survey.

Licensees may use characterization survey data to support the final radiological survey, as long as they can demonstrate that nonimpacted areas at the site have not been adversely affected by decommissioning operations and that the characterization survey data are of sufficient scope and detail to meet the "Information to Be Submitted" guidance for a final survey.

#### *4.2.3.1 Acceptance Criteria*

##### *4.2.3.1.1 Regulatory Requirements*

10 CFR 30.36(g)(4)(i), 40.42(g)(4)(i), 70.38(g)(4)(i), and 72.54(g)(1)

##### *4.2.3.1.2 Regulatory Guidance*

NUREG-1575, "Multi-Agency Radiological Survey and Site Investigation Manual (MARSSIM)"

##### *4.2.3.1.3 Information to Be Submitted*

The information supplied by the licensee should be sufficient to allow the NRC staff to determine whether the characterization survey design is adequate to assess the radiological status of the facility. The licensee should describe the radiation characterization survey design and the results of the survey, including the following:

- a description and justification of the survey measurements for affected media (for example, building surfaces, building materials (volumetric contamination) contamination, surface soil, subsurface soil, surface water, groundwater, sediments, as appropriate)

- a description of the field instruments and methods that were used for measuring concentrations and the sensitivities of those instruments and methods
- a description of the laboratory instruments and methods that were used for measuring concentrations and the sensitivities of those instruments and methods
- the survey results, including tables or charts of the concentrations of residual radioactivity measured
- maps or drawings of the site, area, or building showing areas classified as nonimpacted or impacted and visually summarizing residual radioactivity concentrations in impacted areas
- a justification for classifying areas as nonimpacted
- a discussion of why the licensee considers the characterization survey to be adequate to demonstrate that it is unlikely that significant quantities of residual radioactivity have gone undetected
- a discussion of how areas or surfaces in a survey unit were surveyed or why they did not need to be surveyed if considered to be inaccessible or not readily accessible
- for sites, areas, or buildings with multiple radionuclides, a discussion justifying the ratios of radionuclides that will be assumed in the FSS or an indication that no fixed ratio exists, and each radionuclide will be measured separately (note that this information may be developed and refined during decommissioning, and licensees may elect to include a plan to develop and justify final radionuclide ratios in the DP)

The information to be submitted is also included as part of the DP Checklist provided in this NUREG report (see Section XIV.b from Appendix D of Volume 1).

Licensees should note that, if they elect to dispose of buildings and structures rather than leave them in place (for unrestricted release), the LTR does not apply to the material moved offsite from those buildings and structures. Rather, building and structure deconstruction and dismantlement materials can be released from the site in accordance with existing license conditions. The data from the characterization survey may be sufficient to demonstrate compliance with the conditions of the existing license for releasing material from the site. However, a characterization survey may not be required to demonstrate compliance with the license condition for releasing material from the site. Section G.2.1 of Appendix G of this volume provides additional guidance on the offsite disposition of materials.

#### *4.2.3.2 Evaluation Criteria*

The NRC staff should verify that the licensee has adequately characterized the site, area, or building relative to the location and extent of residual radioactivity. An adequate characterization is one that permits planning for a remediation that will be effective and will not endanger the remediation workers, demonstrates that it is unlikely that significant quantities of residual radioactivity have gone undetected, and provides information that will be used to design the FSS. The extent of detail in the information provided by the licensee should be appropriate for the specific site, area, or building.

The NRC staff should verify that the survey design and results demonstrate that the licensee or responsible party has adequately characterized the site. The characterization survey is adequate if it meets the criteria in the following guidance:

- Section 5.3 of MARSSIM for the characterization survey (the NRC staff may use the "Example Characterization Survey Checklist" in Section 5.3 of MARSSIM for evaluating the licensee's submittal)
- MARSSIM Chapter 6 and Appendix E for instrument capabilities and sensitivities
- MARSSIM Section 4.8.4 for preparing areas for survey

## 4.3 Remedial Action Support Surveys

The effectiveness of remediation efforts in reducing residual radioactivity to acceptable levels is monitored by a remedial action support survey as the remediation effort is in progress. This type of survey activity guides the cleanup in a real-time mode; it also ensures that the remediation workers, the public, and the environment are adequately protected against exposures to radiation and radioactive materials arising from the remediation activities.

The remedial action support survey typically provides a simple radiological parameter, such as direct radiation near the surface being remediated. The level of radiation, below which there is reasonable assurance that the guideline values have been attained, is determined and used for immediate, infield decisions. Such a survey is intended for expediency and does not provide thorough or accurate data describing the final radiological status of the site.

The remedial action support survey is applicable to monitoring surfaces and soils or other bulk materials only if the radionuclides of concern are detectable by field survey techniques. For radionuclides and media that cannot be evaluated at guideline values by field procedures, samples are to be collected and analyzed to evaluate the effectiveness of remediation efforts. For large projects, the use of mobile field laboratories can provide more timely decisions on the effectiveness of remedial actions. Examples of situations for which remedial action support surveys would not be practicable are (1) when soil contains pure alpha or beta emitting radionuclides and (2) when very low energy beta emitters such as tritium are present on surfaces.

Licensees conduct remedial action support surveys to do the following:

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

The determination that a survey unit is ready for an FSS following remediation is an important step in the RSSI process. Remedial activities may result in changes to the distribution of residual radioactivity within the survey unit. Thus, for many survey units, the site-specific parameters used during FSS planning (e.g., variability in the radionuclide concentration, probability of small areas of elevated activity) may need to be confirmed or reestablished following remediation. Obtaining updated values for these critical parameters should be considered when planning a remedial action support survey. In some cases, where

concentrations of some radionuclides after remediation may be very low, it may be useful for licensees to show that certain radionuclides can be considered insignificant; in that case, further detailed evaluation as part of the FSS may not be necessary (see Section 3.3 of this volume). However, the dose from the insignificant radionuclides must be accounted for in demonstrating compliance with the applicable dose criteria.

Note that this survey does not provide information that can be used to demonstrate compliance with the DCGLs and is an interim step in the compliance demonstration process. The FSS will then survey in detail areas that are likely to satisfy the DCGLs on the basis of the remedial action support survey. Alternatively, the remedial action support survey can be designed to meet the objectives of an FSS.[2] DCGLs may be recalculated, based on the results of the remediation process, although a license amendment may be needed to change (increase) previously approved DCGLs in a DP or LTP.

### 4.3.1 Areas of Review

Staff review of the description of the remedial action support surveys should verify that the licensee has designed these surveys appropriately and to assist in determining when remedial actions have been successful, so that it may begin the FSS. In addition, information from these surveys may be used to provide the principal estimate of residual radioactivity variability that will be used to calculate the FSS sample size in a remediated survey unit.

#### *4.3.1.1 Acceptance Criteria*

##### *4.3.1.1.1 Regulatory Requirements*

10 CFR 30.36(g)(4)(ii), 40.42(g)(4)(ii), and 70.38(g)(4)(ii),

##### *4.3.1.1.2 Regulatory Guidance*

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

##### *4.3.1.1.3 Information to Be Submitted*

The NRC staff should verify the licensee’s or responsible party’s description of the support survey includes the following information:

- a description of field screening methods and instrumentation
- a demonstration that field screening should be capable of detecting residual radioactivity at the $DCGL_W$

The information to be submitted is also included as part of the DP Checklist provided in this NUREG report (see Section XIV.c from Appendix D of Volume 1).

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[2] In certain cases, it may be prudent to collect data of sufficient quality during remediation to support the final status survey during remediation. For example, for large subsurface soil excavations, it may be more practical to collect samples at the bottom and sides of the excavation and/or perform scanning prior to filling in the excavation. See Section G.3.2 for additional details.

*4.3.1.1.4 Evaluation Criteria*

The NRC staff should verify that the description of the remedial action support surveys meets (1) the criteria in MARSSIM Section 5.4 for performing remedial action support surveys and (2) the criteria in the applicable MARSSIM chapters listed in this volume for evaluating technical issues, such as appropriate surveys instruments and survey instrument sensitivity.

## 4.4 Final Status Survey Design

Professional judgment and biased sampling are important for locating residual radioactivity and characterizing the extent of residual radioactivity at a site. However, the MARSSIM focus is on planning the FSS, which uses a systematic approach to sampling. Systematic sampling is based on rules that try to achieve the representativeness assumed by the statistical tests.

The licensee uses the FSS to demonstrate compliance with regulations. The primary objectives of the FSS are to do the following:

- verify survey unit classification
- demonstrate that the potential dose from residual radioactivity is below the release criterion for each survey unit
- demonstrate that the potential dose from small areas of elevated activity is below the release criterion for each survey unit

Data provided by the FSS can demonstrate that all radiological parameters satisfy the established guideline values and conditions.

### 4.4.1 Areas of Review

The purpose of the NRC staff's review is to verify that the FSS design is adequate to demonstrate compliance with the radiological criteria for license termination.

*4.4.1.1 Acceptance Criteria*

*4.4.1.1.1 Regulatory Requirements*

10 CFR 20.1501(a), 30.36(g)(4)(iv), 40.42(g)(4)(iv), 70.38(g)(4)(iv), and 72.54(g)(4)

*4.4.1.1.2 Regulatory Guidance*

- Draft NUREG-1505, Rev. 1, "A Nonparametric Statistical Methodology for the Design and Analysis of Final Status Decommissioning Surveys – Interim Draft Report for Comment and Use"
- NUREG-1575, "Multi-Agency Radiological Survey and Site Investigation Manual (MARSSIM)"
- NUREG-1507, "Minimum Detectable Concentrations with Typical Survey Instruments for Various Contaminants and Field Conditions"

#### *4.4.1.1.3 Information to Be Submitted*

The information supplied by the licensee should be sufficient to allow the NRC staff to determine whether the FSS design is adequate to demonstrate compliance with the radiological criteria for license termination. The information should include all of the following:

- a brief overview describing the FSS design
- a description and map or drawing of affected areas of the site, area, or building classified by residual radioactivity levels (Class 1, 2, or 3) and divided into survey units, with an explanation of the basis for division into survey units (maps should have compass headings indicated)
- a description of the background reference areas and materials, if they will be used, and a justification for their selection
- a summary of the statistical and other tests that will be used to evaluate the survey results, including the elevated measurement comparison (EMC), if Class 1 survey units are present; a justification for any test methods not included in MARSSIM; and the values for the decision errors (α and β) with a justification for α and β values greater than 0.05 for Scenario A and B, respectively
- a description of scanning instruments, methods, calibration, operational checks, coverage, and sensitivity for each media and radionuclide
- a description of the instruments, calibration, operational checks, sensitivity, and sampling methods for *in situ* sample measurements, with a demonstration that the instruments and methods have adequate sensitivity (noting that if a licensee uses an advanced technology (e.g., *in situ* gamma spectroscopy), it must be shown to perform with sensitivities that allow detection of residual radioactivity at an appropriate fraction of the DCGL and corresponding investigation levels (ILs))
- a description of the analytical instruments for measuring samples in the laboratory, including the calibration, sensitivity, and methodology for evaluation, with a demonstration that the instruments and methods have adequate sensitivity
- a description of how the samples to be analyzed in the laboratory will be collected, controlled, and handled
- a description of the FSS ILs and how they were determined

The information to be submitted is also included as part of the DP Checklist provided in this NUREG report (see Section XIV.d from Appendix D of Volume 1). Appendix A provides additional information about demonstrating the appropriate selection of survey instrumentation.

#### *4.4.1.2 Evaluation Criteria*

The NRC staff review should verify that the FSS design is adequate to demonstrate compliance with the radiological criteria for license termination. The FSS design is adequate if it meets the criteria in the following guidance:

- Appendix A to this volume, for general guidance on implementing the MARSSIM approach for conducting FSSs
- Appendix B to this volume, for guidance on alternative methods of FSS for simple situations
- MARSSIM Sections 4.4 and 4.6 for classifying areas by residual radioactivity levels and dividing areas into survey units of acceptable size
- MARSSIM Section 4.5 for methods to select background reference areas and materials
- NUREG-1505, Chapter 13, for a method to account for differences in background concentrations between different reference areas
- MARSSIM Section 5.5.2 for statistical tests
- Appendix A to this volume, Section A.8.2, for decision errors
- MARSSIM Sections 6.5.3 and 6.5.4 for selection of acceptable survey instruments, calibration, and operational checkout methods
- MARSSIM Section 6.7 for methods to determine measurement sensitivity; NUREG-1507 for instrument sensitivity information
- MARSSIM Sections 5.5.2.4, 5.5.2.5, 5.5.3, 7.5, and 7.6 for scanning and sampling
- MARSSIM Section 7.7 for sample analytical methods (Table 7.2 in Section 7.7 for acceptable analytical procedural references)
- MARSSIM Sections 7.5 and 7.6 for methods for sample collection
- MARSSIM Section 5.5.2.6 for survey ILs
- Appendix G to this volume for surveys for special structural or land situations

## 4.5 Final Status Survey Report

To the extent possible, the FSSR should stand on its own with minimal information incorporated by reference. Although the FSS is discussed as if it were an activity performed at a single stage of the site investigation process, this does not have to be the case. Data from other surveys conducted during the RSSI process—such as scoping, characterization, and remedial action support surveys—can provide valuable information for an FSS, provided the data are of sufficient quality.

### 4.5.1 Areas of Review

The purpose of the NRC staff review is to verify that the results of the FSS demonstrate that the site, area, or building meets the radiological criteria for license termination. For licensees who have submitted a DP, the FSSR need only include the information described under Section 4.5.1.1 (Acceptance Criteria). A licensee who has not submitted a DP should contact

the NRC staff to ensure its FSSR includes not only the information below but also any other relevant information the staff needs to carry out its review.

#### *4.5.1.1 Acceptance Criteria*

##### *4.5.1.1.1 Regulatory Requirements*

10 CFR 20.1402, 20.1403, 20.1501, 30.36(j)(2), 40.42(j)(2), 70.38(j)(2), and 72.54(l)(2)

##### *4.5.1.1.2 Regulatory Guidance*

NUREG-1575, "Multi-Agency Radiological Survey and Site Investigation Manual (MARSSIM)"

##### *4.5.1.1.3 Information to Be Submitted*

The information submitted by the licensee should be sufficient to allow the staff to determine whether the site, area, or building meets the radiological criteria for license termination. The information should include the following:

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

- a discussion of anomalous data, including any areas of elevated direct radiation detected during scanning that exceeded the IL or any measurement locations in excess of $DCGL_W$

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

- a description of any changes in initial survey unit assumptions relative to the extent of residual radioactivity (e.g., material not accounted for during site characterization)

- a description of how ALARA practices were employed to achieve final activity levels

The information to be submitted is also included as part of the DP Checklist provided in this NUREG report (see Section XIV.e from Appendix D of Volume 1).

*4.5.1.2 Review Procedures*

After review of the FSSR, the NRC reviewer should have reasonable assurance that the FSSR demonstrates that residual radioactivity at the facility complies with the criteria of 10 CFR Part 20, Subpart E. The following guidance discusses the minimal review that should be performed and how the reviewer should select survey units for more detailed reviews.

Section 4.5.1.1.3 describes the minimum information to be submitted in each FSSR. Additional information about the recommended level of documentation is in Appendix D of this volume. At individual facilities, the NRC reviewer many need additional information on site-specific issues and complex technical topics to evaluate the FSSR. In addition, the NRC reviewer may need to obtain previous NRC-generated reports on the FSS, including but not necessarily limited to inspections, confirmatory surveys, and any safety evaluation reports that may have addressed the FSS plan.

*4.5.1.2.1* Minimal Technical Review

The NRC reviewer should review all of the following:

- the results of previously conducted in-process inspections and confirmatory surveys to verify that the licensee has properly implemented the FSSP and associated procedures

- the licensee's QA/QC program, if it has not been previously reviewed

- changes made to the DP or LTP, if not previously reviewed, to confirm that the changes are not significant and are technically correct

- specific parts of the FSS and supporting data that affect the FSS that were not available when the DP or LTP was approved (data that may include supplemental characterization results, basis for final surrogate ratios for multiple radionuclides, or other data collected to specifically support the FSS)

- issues (1) identified by interveners and stakeholders and (2) raised in allegations, to ensure such issues have been satisfactorily resolved

- descriptions of the survey units, to determine if any special survey situations are present (see Appendix G of this volume for examples)

- results of elevated measurement comparisons, to confirm that small areas of residual radioactivity do not exceed the appropriate limits (e.g., $DCGL_{EMC}$) and

- the results of the appropriate statistical tests (e.g., Wilcoxon Rank Sum (WRS) and sign tests), to confirm that results indicate compliance

The purpose of the NRC staff review of in-process inspections, confirmatory surveys, and licensee procedures is to ensure all of the following:

- the FSSs were implemented in accordance with the approved FSSP

- judgmental survey results are not used in the statistical tests and are evaluated separately against the release criteria, and survey results obtained using random start and systematic sampling are statistically treated separately for the purpose of demonstrating compliance

- the QA/QC program was adequate and implemented for the FSS

- inadequacies in the FSS design or implementation were corrected (e.g., the licensee improved the overall FSS design and implementation, using information from survey units for which the release criteria were not initially met and resurvey or further remediation was needed)

- results of confirmatory surveys, including split samples or independent measurements, are consistent with results of licensee surveys

- appropriate instrumentation, with sufficient sensitivities, proper calibrations, and adequately trained users, was used for surveys, scans, and measurements, as described in the FSSP

#### *4.5.1.2.2 Detailed Technical Review*

Along with the minimal review described, the NRC reviewer may perform detailed reviews for a number of survey units. The number initially chosen for detailed review should use a risk-informed approach and the results of the minimal review. The reviewer should consider past inspection history, results of confirmatory surveys, the relative difference between residual radioactivity concentration and the associated DCGLs, the complexity of the FSSP, and the radionuclide mix. The detailed review could include confirming the selection process and location of measurements using survey unit maps or floor plans, checking measurement results using parameters that are specific to the survey methodology, and re-creating the appropriate MARSSIM statistical test results.

#### *4.5.1.2.3 Selecting Survey Units for Detailed Reviews*

Discriminating factors that may be used to select specific survey units for detailed review are listed below. A survey unit that is characterized by one or more of these factors should be considered for potential detailed review. However, the NRC reviewer should focus on survey

units for which there are risk-significant issues, issues that are prevalent across a large number of survey units instead of isolated cases, and issues involving an inadequate basis for conclusions.

These factors include any of the following:

- inconsistencies in defining survey units, including the following:
  - size different from recommended size
  - multiple areas now combined as one larger Class 1 survey unit
  - Class 3 survey units that are bordered by Class 1 units
  - survey units bordered by PSR areas
  - gerrymandered survey unit boundaries
- application of nonstandard statistical tests (e.g., other than WRS test or Sign test)
- significant inconsistencies between the DP/LTP and implemented FSS, including the following examples:
  - use of surface and detector efficiencies that do not match survey methods, surface features, and instrumentation used
  - type of survey instrumentation
  - sample collection method
  - laboratory analytical methods
  - any survey unit where the scan coverage is less than 100 percent for Class 1 areas or less than the minimum commitment for Class 2 or 3 areas
  - number of samples per survey unit
- survey units that were remediated
- survey units for which confirmatory surveys had results inconsistent with the licensee's FSS results
- any Class 2 survey unit with final measurement results near the DCGLW (e.g., greater than 75 percent) or any Class 3 survey unit with significant residual radioactivity (e.g., concentrations greater than 10–25 percent of the DCGLW)
- any survey unit that was downgraded in classification (i.e., from Class 1 to 2, Class 2 to 3, or Class 1 to 3, or from impacted to nonimpacted)
- units surveyed before resolution of QA/QC concerns

- significance of the variability in concentrations (i.e., heterogeneity) across survey units
- inconsistent approach or inadequate basis for determining surrogate radionuclide ratios
- significant changes to DP or LTP that affect the FSS or that were not previously reviewed
- reclassification schemes not approved by the NRC staff
- use of MARSSIM survey methods and statistical tests when hot particles are present
- presence of systems and components, buried and embedded piping, or building foundations slated to remain on the site after license termination
- survey units that combine, for demonstrating compliance, the results of random start or systematic sampling patterns with biased or judgmental survey results
- a survey unit that involves surveying or sampling of media other than building surfaces and surface soils (e.g., groundwater, surface water, sediments, or deep subsurface soils)
- survey units with areas that are hard to access or have abnormal geometries
- any survey unit that combines survey results with a dose assessment or area factors to demonstrate compliance (mixed approaches are used)
- the use of composite sampling to establish compliance with release criteria (i.e., DCGLs). Additional information on composite sampling can be found in Appendix O of this volume.

#### *4.5.1.2.4 Detailed Review Topics*

The detailed review could include confirming the selection process and location of measurements using survey unit maps or floor plans, checking measurement results using parameters that are specific to the survey methodology, and re-creating the appropriate MARSSIM statistical test results. In performing detailed reviews, reviewers should consider, but not necessarily be limited to the following questions:

- Does the FSSR adequately address the issues previously discussed under the selection criteria for detailed reviews, immediately above? For example, if a survey technique was changed from the approved technique, did the FSSR adequately justify the new technique?
- Are the probabilities of Type I and Type II errors acceptable?
- Does the licensee's analysis rely on a large number of results expressed at minimum detectable activity or MDC values?
- Are all of the static measurement or sampling locations for a survey unit taken from a single random-start sampling set, without substitution (e.g., in cases where additional remediation was performed)?

- Is there a discernible trend in results within and among survey units (e.g., when comparing survey methods, locations, or media matrices)?
- If there are discernible trends in the results, are the statistical tests appropriate?
- Are there any outliers in the data? How were they detected and was the disposition of outliers appropriate?
- Are there any assumptions about the variability (variance) of the population?
- What analytical tools (statistical software packages) were used to analyze the data?
- What is the format of the presentation of results? Is it consistent for the survey units reported? For example, are the measurement units consistent with the survey data, the media measured, and the DCGLs?

The detailed review of the initially selected survey units may indicate issues that are prevalent across many units instead of isolated cases. In this case, the reviewer may decide to evaluate additional survey units in detail.

#### *4.5.1.3 Evaluation Criteria*

The NRC review should determine whether the FSSR is adequate to demonstrate compliance with the radiological criteria for license termination. The reviewer should verify that the licensee's FSS results support the conclusion that each survey unit meets the radiological criteria for license termination. The FSS is adequate if it meets the following criteria:

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

## 4.6 Issues not Covered in MARSSIM

MARSSIM's main focus is on providing guidance for the design of the FSSs for residual radioactivity in surface soils and on building surfaces and evaluating the collected data. However, several issues related to releasing sites are beyond the scope of MARSSIM. MARSSIM does not provide guidance for translating the release criterion into DCGLs. MARSSIM can be applied to surveys performed at vicinity properties—those not under licensee control—but the decision to apply MARSSIM at vicinity properties is outside the scope of MARSSIM. MARSSIM does not address other media (e.g., subsurface soil, volumetrically contaminated building materials, groundwater, surface water, sediments) containing residual radioactivity. Nor does it address the disposition of components and equipment that are not part of the survey unit. Some of the reasons for limiting the scope of the guidance to surface soils and building surfaces include (1) residual radioactivity is limited to these media for many sites following remediation, (2) since many sites have surface soil and building surfaces as the leading sources of residual radioactivity, existing computer models used for calculating the concentrations based on dose or risk generally consider only sources associated with surface

soils or building surfaces, and (3) MARSSIM was written in support of cleanup rulemaking efforts for which supporting data are mostly limited to residual radioactivity in surface soils and on building surfaces. Table 4.2 summarizes the scope of MARSSIM. Although this table was taken from MARSSIM, it has been modified to be specific to the needs of NRC licensees.

This volume contains guidance for some topics beyond the scope of MARSSIM. Appendix F has guidance specific to the characterization of groundwater, surface water, and sediments. Chapter 5 and Appendices H, I, J, K, L, M, and Q contain other guidance pertaining to dose modeling. Guidance can be found in Appendix G for special characterization and survey issues such as subsurface residual radioactivity, embedded piping, sewer systems, and paved areas.

## Table 4.2 Scope of MARSSIM

| Within Scope of MARSSIM | Beyond Scope of MARSSIM |
|---|---|
| *Guidance* MARSSIM provides technical guidance on conducting radiation surveys and site investigations. | *Regulation* MARSSIM does not establish new regulations or address nontechnical issues (e.g., legal or policy) for site cleanup. Release criteria will be provided rather than calculated using MARSSIM. |
| *Tool Box* MARSSIM can be thought of as an extensive tool box with many components—some within the text of MARSSIM, others by reference. | *Tool Box* Many topics are beyond the scope of MARSSIM, including public participation programs, packaging and transportation of wastes for disposal, remediation and stabilization techniques, and training. |
| *Measurement* The guidance given in MARSSIM is performance-based and directed toward acquiring site-specific data. | *Procedure* The approaches suggested in MARSSIM vary depending on the various site data needs—there are no set procedures for sample collection, measurement techniques, storage, or disposal established in MARSSIM. |
| *Modeling* The interface between environmental pathway modeling and MARSSIM is an important survey design consideration addressed in MARSSIM. | *Modeling* Environmental pathway modeling and ecological endpoints in modeling are beyond the scope of MARSSIM. |
| *Soil and Buildings* The two main media of interest in MARSSIM are surface soil and building surfaces with residual radioactivity. | *Other Media* MARSSIM does not cover other media, including subsurface soil, surface or subsurface water, biota, air, sewers, sediments, or volumetric building residual radioactivity.<br>*Materials or Equipment* MARSSIM does not cover disposition of materials (including construction materials) or equipment (see Appendix G, Section G.2.1, of this volume). |
| *Final Status Survey (FSS)* The focus of MARSSIM is on the FSS, as this is the deciding factor in judging if the site meets the release criterion. | *Other Survey Types* Although not the focus, MARSSIM provides less detailed information on scoping, characterization, and remedial action support surveys. |
| *Radiation* MARSSIM only considers radiation-derived hazards. | *Chemicals* MARSSIM does not cover any hazards posed by chemical contamination. |
| *Remediation Method* MARSSIM assists in determining when sites are ready for an FSS and provides guidance on how to determine if remediation was successful. | *Remediation Method* MARSSIM does not discuss selection and evaluation of remedial alternatives, public involvement, legal considerations, or policy decisions related to planning. |
| *DQO Process* MARSSIM presents a systemized approach for designing surveys to collect data needed for making decisions such as whether to release a site. | *DQO Process* MARSSIM does not provide prescriptive or default values of DQOs. |
| *DQA* MARSSIM provides a set of statistical tests for evaluating data and lists alternative tests that may be applicable at specific sites. | *DQA* MARSSIM does not prescribe a statistical test for use at all sites. |

# 5 DOSE MODELING EVALUATIONS

## 5.1 Introduction

Decommissioning plans typically include estimates of the potential future dose that could be caused by the residual radioactivity remaining on the site after decommissioning activities are completed. Calculating potential doses allows both the licensee and regulator to take site-specific information into account in determining acceptable concentrations of residual radioactivity at the site using dose models and exposure scenarios that are as realistic as necessary for the given facility. This section has been written to maintain this flexibility. It includes the evaluation findings and supporting detailed technical guidance necessary to review the licensee's dose and ALARA analyses. The discussion on decommissioning groups in Volume 1 of this NUREG series provides guidance on information to be submitted.

Dose modeling information is typically submitted as part of a DP or LTP, although in some cases it may be submitted separately or as part of a FSSR or other document. This chapter usually refers to DPs, although other types of reports are implied, if appropriate. The NRC staff should review all of the dose modeling information submitted by the licensee. For certain cases, such as screening analyses using default values or a lookup table, most of the review has already been completed in developing these tools and, therefore, the licensee need only submit minimal site information and justification in using these models, parameters, and exposure scenarios. In addition, the NRC staff should review the ALARA analyses, which are based, in part, on the dose modeling. Two general approaches exist to provide reasonable assurance that the final concentrations should meet the requirements of Subpart E:

- The licensee can commit to the exposure scenario(s), model(s), and parameters to be used to evaluate compliance with the dose criterion using the final concentrations. The licensee should project expected final concentrations in the DP to show that there is reasonable assurance that the dose criterion will be met at the time of license termination.

- The licensee can derive and commit to meeting nuclide-specific concentration limits equivalent to the dose limit.

The "Decommissioning and License Termination Framework" (Figure 1.2), which generalizes the entire decommissioning process (e.g., Step 7 includes FSS and other requirements related to license termination), provides licensees with guidance on how to perform iterative dose analyses. The NRC staff review of dose modeling consists of evaluations in four general areas:

- the source and source release assumptions

- an exposure scenario considering the site environment

- the mathematical model/computational method used

- the parameter values and a measure of their uncertainty

The actions taken as part of the loop suggested by Steps 8 through 12 of Figure 1.2 can result in the licensee modifying one or more of the above four parts. Licensees, generally, should not

and do not need to provide information on dose modeling iterations that are not the final dose analyses.

In some cases, licensees may wish to include the iterative process as part of the DP. This is, generally, because site characterization is not initially complete enough to provide reasonable justification for assumptions used in modeling the site. Usually, such incorporation would be in the form of license conditions that need to be satisfied before license termination can occur.

For example, a site may have initial data on groundwater contamination but does not currently have enough data on hydrological conditions to determine which survey units will be affected by the plume. Based on the limited data available, the licensee designates an area around the plume, and all survey units that involve that area will include the dose from the groundwater as part of the overall dose analyses. For the purposes of this example, the NRC could require the licensee, through a license condition (or other mechanism), to continue to characterize its groundwater. If the information confirms that the area affected by the groundwater contamination is the same or smaller than the assumed area, the licensee can proceed with the decommissioning process. If the licensee wishes to take advantage of the smaller area, or the data points to a larger affected area, the licensee may need to submit a license amendment request to modify the FSSP, the dose modeling, and any other area of the DP affected by the new assumed groundwater contamination-affected area (e.g., adding or subtracting survey units from the list that would consider groundwater contributions in complying with Subpart E).[1]

As described by Figure 1.2 and the preceding example, the areas of dose modeling, site characterization, and FSSP are interdependent. This is an advantage as judicious use of dose modeling can help guide site characterization. In addition, both site characterization and FSSP can guide development of reasonable exposure scenarios or modeling approaches. For example, the appropriate survey techniques may require more advanced modeling in some areas to make them cost effective to implement.

This chapter and the associated appendices use different terms describing exposure scenarios. Table 5.1 includes a description and comparison of these exposure scenario terms.

[1] Licensees should also consider how groundwater transport may affect other survey units or environmental media (e.g., surface water) and appropriately consider uncertainty in the temporal and spatial distribution of radioactivity in the environment. Source remediation may be an option to reduce future downgradient impacts if the future impacts are found to be unacceptable. Because dispersion and dilution are expected to lead to a decrease in peak concentrations away from the source, calculations performed for the source area, if they capture the peak concentrations, may be used to bound the impacts associated with other areas of the site.

**Table 5.1 Comparison and Description of Exposure Scenario Terms Used in this Guidance**

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

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

## 5.2 General Approach for Dose Modeling

The following section discusses the basic components that are involved in a dose modeling assessment. It is meant to provide an overview of how the pieces fit together. This section should give both licensees and reviewers a high-level understanding of the "big picture" review of the dose assessment evaluation. Section 5.3 provides additional details regarding the information that should be submitted with a DP or LTP request, Section 5.4 provides additional detail regarding the acceptance review performed by NRC staff upon receipt of the request, Section 5.5 provides details related to the safety evaluation review, Section 5.6 summarizes review criteria, and Section 5.7 provides a list of additional guidance documents for use in the preparation of the request by the licensee as well as in the review by NRC staff's review.

Chapter 4 of this volume addresses characterization of the residual radioactivity currently present at the site and radiological surveys. The information is based on measurements and knowledge of the site history. To perform dose modeling, the licensee should use the site information on residual radioactivity expected to be present at the completion of decommissioning to develop a generalized view of the site's expected final source configuration[3]. In developing the source term[4] model, the licensee should consider the site measurements, the intended remedial actions, and the needs of both the conceptual model and the FSS.

For example, a site may have a large number of both historical and current measurements characterizing the residual radioactivity over a 10-hectare (25-acre) site. If the site information shows that residual radioactivity levels do not vary significantly, the licensee may assume that the source is a uniform layer of residual radioactivity over the site. If the site information shows significant variability in residual radioactivity concentration, then the licensee may conceptualize the site as two or more sources of residual radioactivity:

- one or more "hot spot" sources that represent the area(s) of elevated concentration
- a source that represents the larger area of residual radioactivity, which contains residual radioactivity at a lower concentration compared to the elevated area

After the source configuration has been determined, the question becomes, "How could humans be exposed either directly or indirectly to residual radioactivity?" or "What is the appropriate exposure scenario?" Each exposure scenario should address the following scenario questions:

- How does the residual radioactivity move through the environment?
- Where can humans be exposed to the environmental concentrations?

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3 Source configuration refers to the geometry of the source (e.g., shape, including thickness), as well as the distribution of residual radioactivity (e.g., homogenous versus nonhomogeneous).

4 The source term characterizes the release rate of radionuclides from the source zone. The source term is a function of the inventory and the release mechanism (e.g., solubility controlled, desorption, or diffusion). RESRAD-ONSITE, RESRAD-BUILD, and DandD have built-in release mechanisms and models, while RESRAD-OFFSITE offers several options to define the "source term." Section 5.5 contains more information. Note that the definition of source term in this volume is slightly different than the definition of source term found in the NRC glossary. The definition found in the online NRC glossary, at https://www.nrc.gov/reading-rm/basic-ref/glossary/source-term.html, is specific to accidents involving radioactive materials: "types and amounts of radioactive or hazardous material released to the environment following an accident".

- What is/are the likely land use(s) in the future for these areas?

- What are the exposure group's habits that will determine exposure? (What do they eat and where does it come from? How much? Where do they get water and how much? How much time do they spend on various activities?)

In most situations, there are numerous possible exposure scenarios in which future human exposure groups could interact with residual radioactivity. The compliance criteria in 10 CFR Part 20 for decommissioning does not require an investigation of all (or many) possible exposure scenarios; its focus is on the dose to members of the critical group for the compliance exposure scenario. The critical group is defined (at 10 CFR 20.1003, "Definitions") as "the group of individuals reasonably expected to receive the greatest exposure to residual radioactivity for any applicable set of circumstances." The compliance exposure scenario is the exposure scenario that leads to the largest peak dose to the average member of the critical group from the mixture of radionuclides. It may be based on a bounding exposure scenario, a screening exposure scenario, another exposure scenario using conservative assumptions about land uses or behaviors, or a scenario considering reasonably foreseeable future land uses for the area.

If the licensee bases its compliance exposure scenario on reasonably foreseeable land use scenarios which are not clearly bounding, the licensee should also identify less likely but plausible land use scenarios. These are scenarios that could lead to higher doses compared to the reasonably foreseeable land use scenario used to demonstrate compliance with the LTR criteria. The evaluation of less likely but plausible exposure scenarios ensures that, if land uses other than the reasonably foreseeable land use were to occur in the future, unacceptably high risks would not result.

By combining knowledge about the sources of residual radioactivity and the exposure scenario questions, the analyst can develop exposure pathways. Exposure pathways are the routes that residual radioactivity may take in traveling from the source, through the environment, to the receptor or human. Exposure routes can be fairly simple and direct (e.g., residual radioactivity in surface soil emits gamma radiation, which results in direct exposure of an individual standing on the soil), or exposure routes can be fairly complex and indirect (e.g., residual radioactivity in the surface soil leaches to unsaturated soil layers and is transported to the underlying aquifer and water from the aquifer is extracted for use as drinking water, which results in exposure to individuals ingesting groundwater). Exposure pathways typically fall into three principal categories identified by the manner in which the exposed individual interacts with residual radioactivity present in environmental media: ingestion, inhalation, and external radiation exposure.

The exposure pathways for many of the exposure groups can be bounded by a smaller number of possible exposure groups. For example, at a rural site with surface soil residual radioactivity, two possible exposure groups are (1) a gardener who grows a small fraction of his or her fruits and vegetables in the soil and (2) a resident farmer who grows a larger fraction of his or her own food (i.e., the site supplies not only vegetables but also meat and milk). In this case, the resident farmer scenario could bound the gardener exposure scenario because the exposure

pathways and specific parameter values associated with the gardener scenario are bounded by the considerations for the resident farmer.[5]

As required by 10 CFR 20.1402, "Radiological criteria for unrestricted use," expected doses are evaluated for the average member of the critical group, whose characteristics differ from those of the maximally exposed individual. This is not a reduction in the level of protection provided to the public but is an attempt to emphasize the uncertainty and assumptions needed in calculating potential future doses while limiting boundless speculation on possible future exposure scenarios. While it is possible to actually identify, with confidence, the most exposed member of the public in some operational situations (e.g., through monitoring, time-studies, distance from the facility), identifying the specific individual who should receive the highest dose some time (up to 1,000 years) in the future 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 the statistical results from data sets (i.e., 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 (e.g., the maximum amount of meat, milk, vegetables, possible exposure time), the result could be an extremely conservative calculation of an unrealistic exposure scenario and could lead to excessively low allowable residual radioactivity levels.

Calculating the dose to the critical group is intended to bound the individual dose to other possible exposure groups. The critical group is a relatively small group of individuals who, due to their habits, actions, and characteristics, could receive among the highest potential doses at some time in the future. By using the hypothetical critical group as the dose receptor, coupled with prudently conservative models, it is unlikely that any individual would actually receive doses in excess of those calculated for the average member of the critical group. The description of a critical group's habits, actions, and characteristics should be based on credible assumptions, and the information or data ranges used to support the assumptions should be limited in scope to reduce the possibility of adding members of less exposed groups to the critical group. An analysis of the average member of the critical group's potential exposure should also include, in most cases, some evaluation of the uncertainty in the parameter values used to represent physical properties of the environment.

The definitions in 10 CFR Part 20 should be used when calculating dose to demonstrate compliance with the requirements of Subpart E. The intake-to-dose conversion factors from Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," issued September 1988 (EPA 520/1-88-020) (EPA, 1988b), which are based primarily on adults, should be used when calculating internal exposures. As stated in EPA's "Federal Radiation Protection Guidance for Exposure of the General Public," dated December 23, 1994, implementing age and sex dependent limits for the general public would be difficult. Not only would the uncertainty associated with the impact of age and sex be significant, but the detailed consideration of age and sex is not generally necessary.

[5] The statement that the resident farmer scenario bounds the resident gardener scenario is true with respect to plant and animal ingestion or consumption rates and pathways; however, it may not be true with respect to the drinking water or irrigation pathways. For example, higher groundwater pumping rates assumed for the resident farmer may, in certain models, lead to a decrease in groundwater concentrations extracted from a well, due to greater dilution from additional clean water being pulled into the well. Therefore, in certain circumstances, the residential farmer scenario may not bound the residential gardener scenario.

Since age-based dose conversion factors are not being used, the same dose conversion factors are applied to all individuals. Only in rare exposure scenarios will a non-adult individual receive a higher dose (i.e., take in more radioactive material) than an adult individual in a similar exposure scenario. One example is the milk pathway: children generally drink more milk annually than adults. If milk were the only pathway that would expose the individual to a dose, then the child would have a slightly higher dose than the adult. But in most situations, especially ones involving multiple pathways, the total intake of the adult is greater than that of a child. Therefore, for most multiple pathway exposure scenarios, such as screening analyses, the average member of the critical group should usually be assumed to be an adult with the proper habits and characteristics of an adult. As the licensee eliminates pathways or modifies the exposure scenario, the behavior and dietary habits of children may become important. In such cases, the licensees should contact the NRC staff for guidance.

By integrating the exposure scenario, source configuration, and knowledge about the applicable environmental transport routes involved in the exposure pathways, a conceptual model of the features and processes at the site can be created. The conceptual model is a qualitative description of the important environmental transport and exposure pathways and their interrelationships. Abstraction is commonly necessary to translate the concepts of a conceptual model into mathematical terms. Not only is model abstraction necessary, it can be useful to an analyst in explaining complex system behavior by reducing the system to its major components, thereby facilitating communication on the most important components of the system being simulated.

The extent of the details provided in the conceptual model can impact the extent to which the features and processes described in the conceptual model can be translated into mathematical terms. In particular, mathematical modeling of groundwater transport can be challenging for a variety of reasons. The characteristics and features that introduce challenges include, but are not limited to, the following:

- limited availability of site-specific data to describe important features, events, and processes for a specific groundwater system
- limited availability of site-specific data on the source term
- natural heterogeneity of the site and changes in site characteristics over time
- complex interactions of the hydrologic, geologic, physical, and chemical processes associated with the system

Model abstraction for hydrogeological systems may require the expertise of a qualified specialist. In some cases, the geology of an area may initially appear to be very complicated; however, after analysis, it may be determined that most of the geological detail need not be abstracted. For example, a bedrock aquifer layer may include smaller faults that could potentially serve as preferred pathways. If characterization studies indicate that the faults are filled with low permeable minerals and the intermittent nature of the faults prevents flow from being slowed or diverted, these faults may not need to be included in the conceptual model(s) and represented in the mathematical model. If, on the other hand, site characterization shows that the faults transport water more quickly than the rock matrix, flow through the faults should be represented in the mathematical model (e.g., the hydraulic conductivity of the matrix could be increased to represent the impact of small faults).

Going from a conceptual model to a mathematical model involves a number of assumptions and simplifications. For example, one part of a conceptual model of surface soil residual radioactivity involves the leaching of radionuclides through the soil and into the aquifer. In reality, the soil between the surface and the aquifer is usually formed by numerous layers of different types of soils with varying thickness across a site. For the purposes of dose modeling, the conceptual model is more focused on knowing how much activity is entering (and leaving) each major environmental compartment (such as the aquifer) than on precisely predicting the level of activity in the intervening material (e.g., any single soil layer between the surface and the aquifer). Therefore, the mathematical model may view the intervening soil layers as one layer or just a few layers, depending on the difficulty of justifying effective parameters that will mimic the real behavior. Users of off-the-shelf codes should be aware of and consider the appropriateness of the assumptions made in the computer model they are using.

The selection of parameter values (or ranges) for the features, events, and processes of a specific site depends not only on the site conditions and the exposure scenario(s) but also on the computer code (or mathematical model) being used. Nearly any data set will need to be transformed into one appropriate to the situation. This can be as straightforward as generating a sitewide effective soil density value or as complex as converting resuspension factor data into resuspension rates. The NRC has already factored these issues into the data used in the screening analyses, but licensees using site-specific information should justify their values.

The conversion of data into parameter values for use in these analyses requires consideration of uncertainty. In the past, the most common computer codes were deterministic and did not explicitly consider parameter uncertainty. Although it is not always necessary to use a probabilistic code to evaluate parameter uncertainty for site-specific analyses, licensees should provide some discussion of the level of uncertainty in the results and understand the most important factors influencing site-specific parameter values to ensure that there is sufficient information to support the results of the analysis. It should be noted that the type of uncertainty of prime interest to the NRC staff is uncertainty in the physical parameters (e.g., default behavioral and metabolic parameters found in NUREG/CR-5512, Volume 3, can be used with limited justification). Appendix Q includes additional guidance on considering uncertainty in dose modeling.

Licensees using probabilistic dose modeling should use the “peak of the mean” dose for demonstrating compliance with 10 CFR Part 20, Subpart E. Similar to all regulatory guidance, this NUREG report contains one approach for determining compliance with the regulations using probabilistic analyses. Use of “mean of the peaks” is also acceptable for demonstrating compliance. If the “mean of the peaks” dose is significantly higher than the “peak of the mean” dose, then “risk dilution” may be an issue in the probabilistic model. Appendix Q contains more information on risk dilution. If the licensee intends to use any probabilistic approach to calculate DCGLs, it should discuss its planned approach with the NRC staff.

## 5.3 Information to be Submitted

Dose modeling information is typically submitted as part of a DP or LTP, though in some cases it may be submitted separately or as part of an FSSR or other document. This information should include the licensee’s assessment of the potential doses resulting from the residual radioactivity remaining at the end of the decommissioning process, as well as a comparison of the potential doses against radiological criteria for license termination found in 10 CFR Part 20, Subpart E. Information needed for performing reviews of DPs varies depending on the decommissioning group. For certain cases, such as screening analyses using default values or

a look-up table, most of the review has already been completed in developing these tools and, therefore, the licensee need only submit minimal site information and justification in using these models, parameters, and exposure scenarios. On the other hand, site-specific dose modeling associated with Decommissioning Groups 4 – 7 may require extensive details regarding the source term, exposure scenarios, analytical methods, and other details related to the proposed action.

The information to be submitted is also included as part of the DP Checklist provided in this NUREG report (see Checklist Section V.b from Appendix D of Volume 1).

### 5.3.1 Decommissioning Groups 1 – 3 (Unrestricted Release Using Screening Criteria)

For DPs for sites associated with Decommissioning Groups 1 – 3, the licensee should provide a description of the exposure scenarios proposed to show compliance with the regulations. The licensee's dose modeling for building surfaces or surface soil using the default screening criteria should include the general conceptual model (for both the radiological contaminants of concern and the building or outside environment) of the site and a summary of the screening method used (i.e., running DandD[6] or using the lookup tables [see Appendix H]).

The information to be submitted is also included as part of the DP Checklist provided in this NUREG report (see Checklist Section V.a.1. (Building Surface Residual Radioactivity) or V.a.2. ("Surface Soil Residual Radioactivity") from Appendix D of Volume 1).

### 5.3.2 Decommissioning Groups 4 – 5 (Unrestricted Release Using Site-Specific Information)

In addition to providing information on the exposure scenarios, the licensee should also provide information on the parameters used in the site-specific analysis. "Site-specific" is used in a general sense to describe all dose analyses except those based only on the default screening tools. This may be as simple as a few parameter changes from the default values in the DandD computer code to licensees using exposure scenarios, models, and parameter values that are only applicable at the licensee's site. The information submitted should include the following:

- site-specific source term information, including nuclides of interest, configuration of the source, areal variability of the source, release mechanisms, and so forth
- a description of the compliance exposure scenario, including a description of the critical group
- a description of any other reasonably foreseeable or less likely but plausible exposure scenarios considered
- a description of the conceptual model for the specific site, including the source term, the physical features important to modeling the transport pathways, and the critical group

6 Decommissioning codes such as DandD can be obtained from NRC's RAMP (Radiation Protection Computer Code Analysis and Maintenance Program).

- identification, description, and justification of the computer code(s) and/or mathematical model(s) used (e.g., hand calculations, DandD Version 2 (or later version), RESRAD family of codes)

- a description of the parameters and the basis for the parameter values used in the analysis

- a discussion about the effect of uncertainty on the results

- input and output files or printouts, if a computer program was used

The information to be submitted is also included as part of the DP Checklist provided in this NUREG report (see Checklist Section V.b from Appendix D of Volume 1).

### 5.3.3 Decommissioning Group 6 (Restricted Release)

The majority of the information needed for Decommissioning Group 6 is the same as what would be needed for Decommissioning Groups 4 – 5. In addition to the information listed above for Decommissioning Groups 4 – 5, details related to potential impacts associated with a loss of institutional controls and an ALARA analyses should be included.

The information to be submitted is also included as part of the master DP Checklist provided in this NUREG report (see Section V.c from Appendix D of Volume 1).

### 5.3.4 Decommissioning Group 7 (Alternate Release Criteria)

The same information provided for restricted release should also be provided for release using alternate criteria (see Section 5.3.3). Additionally, the licensee should provide information on the basis for alternative criteria, including an ALARA analysis. The types of information provided are included in NUREG-1757, Volume 1, Chapter 17.8 and Appendix M, as well as SECY-03-0069 and Regulatory Issue Summary 2004-08.

The information to be submitted is also included as part of the master DP Checklist provided in this NUREG report (see Section V.d from Appendix D of Volume 1).

## 5.4 Acceptance Review

Upon receipt of a DP, NRC staff should perform a high-level review of the contents of the submittal in an effort to organize the review and evaluation process. The NRC staff should organize the review by first looking at the overall scope of the dose modeling provided (possibly for several decommissioning options or critical groups, or both). This review should determine which decommissioning group and corresponding review criteria should be used and which specific dose modeling sections are applicable for the given DP.

One acceptable way to organize this initial review is (1) to identify and confirm the principal sources of residual radioactivity (before and after remediation) and (2) to identify the decommissioning goal of the DP. Coupling these two sets of information, the NRC staff should have a good indication of what sections of the guidance apply. For decommissioning goals involving unrestricted release, the NRC staff should quickly evaluate the appropriate decommissioning group to which the licensee belongs (i.e., Decommissioning Groups 1-5). Sections 5.4 and 5.5 are organized by review step with subheadings in each section organized

by decommissioning group. Therefore, it is important to determine the appropriate decommissioning group to effectively use the guidance in this chapter.

The NRC staff should confirm that conditions at the site are consistent with the approach chosen by the licensee and the decommissioning group's requirements (e.g., whether the conditions of the site are consistent with the modeling assumptions inherent in the screening analysis approach). A screening approach is generally inappropriate for sites exhibiting any of the following conditions (excluding those caused by sources of background radiation):

- subsurface soil residual radioactivity (screening analyses assume only *surface* soil residual radioactivity)
- radionuclide residual radioactivity present in an aquifer
- buildings with volumetrically contaminated material
- radionuclide concentrations in surface water sediments
- sites that have an infiltration rate that is greater than the vertical saturated hydraulic conductivity (i.e., resulting in the water running off the surface rather than only infiltrating into the ground)

These conditions are inappropriate because they are inconsistent with the conceptual models used in developing the screening values. In other words, the conceptual model, parameters, and exposure scenarios used by the DandD computer code to conduct screening analyses and develop the screening tables presented in Appendix H, are generally incompatible with such conditions. Situations do exist where licensees may still use the screening analyses even if one or more of the conditions listed in the bulleted list above is applicable. For example, by conservatively assuming buried radioactive material is excavated and spread across the surface, the screening criteria may be applicable for use at the site. However, if exceptions are made, it is important to understand the underlying assumptions used to develop the screening values to ensure that the risk is not underestimated (e.g., if screening values are used for buried residual radioactivity it is important to consider the thickness of residual radioactivity and other factors to ensure that the use of surface soil screening values based on 15 cm of residual radioactivity adequately or conservatively represents the risk from the buried residual radioactivity).

As part of the acceptance review, NRC staff should review the DP to ensure that the licensee or responsible party has included the necessary information and determine if the level of detail appears to be adequate for a detailed technical review. This acceptance review should include a general review of the DP table of contents and the individual sections of the submittal. NRC staff should also review the dose modeling portion of the DP but does not need to assess the technical accuracy or completeness of the information contained therein, which should be determined during the detailed technical review if the DP is accepted for review. The NRC staff should also verify that the licensee provided enough information to allow an independent evaluation of the potential dose resulting from the residual radioactivity after license termination and reasonable assurance that the decommissioning option will comply with regulations.

Specific considerations for acceptance reviews vary based on the decommissioning group associated with the site and the proposed path forward. The following subsections provide specific details regarding what should be considered based on whether NRC staff will be

performing a review for a site using screening criteria, a review for sites using site-specific information and seeking unrestricted release, a review for sites using site-specific information and seeking restricted release or a review for sites using site-specific information and seeking to use alternate criteria.

### 5.4.1 Decommissioning Groups 1 – 3 (Unrestricted Release Using Screening Criteria)

Decommissioning sites included in Decommissioning Groups 1 – 3 can be evaluated using screening criteria. Decommissioning Group 1 includes sites in which licensed material was used in a way that precluded the release of radioactivity into the environment and would not be expected to have contaminated areas above the screening criteria. These sites generally include licensees that possessed and used sealed sources. These sites do not require a decommissioning plan and dose modeling is not performed. Decommissioning Groups 2 and 3 include sites where residual radioactive contamination is present on building surfaces and in soils, and licensees are typically able to demonstrate that their facilities meet the provisions of 10 CFR 20.1402 ("Radiological criteria for unrestricted use") using a screening approach. Decommissioning Group 2 sites do not need to submit a DP since specific cleanup activities and procedures consistent with remediating the facility are already included in their current license. Decommissioning Group 3 sites require amendments to their license to incorporate necessary remediation procedures needed to decommission the site. Specific details related to these groups are included in NUREG-1757, Volume 1.

When evaluating a DP for a site in Decommissioning Groups 1 – 3, NRC staff should review the DP to ensure that, at a minimum, it contains the information needed to make conclusions regarding the DP's compliance with 10 CFR 20.1402. For residual radioactivity on surface soils, the residential farmer scenario is considered, while for residual radioactivity on building surfaces, the building occupancy exposure scenario is considered. For screening analyses, the licensee should provide information for NRC staff to evaluate the appropriateness of use of a screening approach (e.g., information on the expected source configuration should be provided to allow staff to evaluate whether screening assumptions related to the source are met). The staff should also perform a high-level review of the dose modeling portion of the DP without assessing the technical accuracy and completeness of the information contained therein. Specific details will be evaluated during the more detailed technical review.

### 5.4.2 Decommissioning Groups 4 – 5 (Unrestricted Release Using Site-Specific Information)

Decommissioning Groups 4 – 5 include sites that are being considered for unrestricted release using site-specific dose analyses. Decommissioning Group 4 includes sites that are not found to include groundwater contamination while Decommissioning Group 5 sites include groundwater contamination. Additional details on these decommissioning groups are included in NUREG-1757, Volume 1.

Considerations when assessing the path forward for decommissioning actions with a goal of terminating the license under the unrestricted release requirements of 10 CFR 20.1402, include the primary exposure scenarios for the individuals exposed on the site. Exposure scenarios should be consistent with reasonably foreseeable land uses over the next decades, or use a bounding exposure scenario, such as a resident farmer. The residential farmer scenario is typically a bounding exposure scenario for residual radioactivity in the environment because this group includes a nearly comprehensive number of exposure pathways (footnote 13 explains how, in certain cases, a residential gardener scenario may lead to higher doses than a

residential farmer scenario). In addition to pathways that may be limited by land use assumptions, site conditions, such as soil type or groundwater quality, may remove potential exposure pathways from consideration with the appropriate level of justification by the licensee. In rare instances, an exposure scenario involving offsite use of residual radioactivity may be the critical exposure scenario (e.g., buried contamination that is transported offsite via surface water where a critical group with unique exposure pathways or subsistence land use could be exposed).

When performing an acceptance review on a DP for a site in Decommissioning Groups 4 and 5, NRC staff should ensure that, at a minimum, information on the source term[7], exposure scenario(s), conceptual model(s), numerical analyses (e.g., hand calculations or computer models), and uncertainty have been included to allow the NRC staff to make conclusions regarding the ability of the licensee to comply with 10 CFR 20.1402. The NRC staff may also perform a high-level review of the assumptions regarding the source term, the conceptual model of the site or building as appropriate, the exposure scenario(s), the mathematical method employed, and the parameters used in the analysis and their uncertainty.

### 5.4.3 Decommissioning Group 6 (Restricted Release)

Decommissioning Group 6 sites are considered for restricted release and therefore conclusions are made regarding the ability of the site to comply with 10 CFR 20.1403 at the time of license termination based on the information provided in the DP. In addition to the information listed above for Decommissioning Groups 4 – 5, the DP should consider, at a minimum, two different sets of exposure scenarios. One set of exposure scenarios should evaluate the dependence on the proposed institutional controls or restrictions by assuming the institutional controls are effective. Depending on where the residual radioactivity is and what the proposed restrictions are, the exposure location(s) for the critical group could be either onsite or offsite. The second set of exposure scenarios should assume that institutional controls are “no longer in effect” in accordance with 10 CFR 20.1403(e) (i.e., institutional controls put in place by the licensee have failed to work properly, or effectively, and that the site will be used without knowledge of the presence of residual radioactivity). Although various times can be evaluated in sensitivity analyses, institutional controls should be assumed to be ineffective immediately after license termination.

Chapter 6 of NUREG-1200, “Standard Review Plan for the Review of a License Application for a Low-Level Radioactive Waste Disposal Facility,” Revision 3, issued in April 1994, can be used as a guideline on the development of site-specific acceptance review criteria for restricted release as applicable (e.g., radioactive release and transport, intruder protection, erosion protection during the post-closure phase, which may be applicable for restricted release scenarios involving offsite receptors or use of engineered barriers). Additionally, Appendix J of this document contains information for use when considering intrusion scenarios for buried radioactivity.

Similar to the acceptance review for Decommissioning Groups 4 – 5, NRC staff should ensure that, at a minimum, information on the source term, exposure scenario(s), conceptual model(s), numerical analyses (e.g., hand calculations or computer models), and uncertainty have been included. The NRC staff may also perform a high-level review of the assumptions regarding the

[7] The source term characterizes the release rate of radionuclides from the source zone. The source term is a function of the inventory and the release mechanism (e.g., solubility-controlled, desorption, or diffusion-limited release). RESRAD-ONSITE, RESRAD-BUILD, and DandD have built-in release models, while RESRAD-OFFSITE offers several options to define the “source term.” Appendix I, Section 2, provides more information.

source term, the conceptual model of the site or building as appropriate, the exposure scenario(s), the mathematical method employed, and the parameters used in the analyses and their uncertainty.

### 5.4.4 Decommissioning Group 7 (Alternate Release Criteria)

Decommissioning Group 7 sites are evaluated using a licensee's proposed alternate criteria. Section 5.4.3 for Decommissioning Group 6 and Chapter 6 of NUREG-1200, "Standard Review Plan for the Review of a License Application for a Low-Level Radioactive Waste Disposal Facility," Revision 3, issued in April 1994, may be used as guidelines on the development of site-specific acceptance review criteria for alternate decommissioning criteria, as applicable.

When evaluating sites in Decommissioning Group 7 NRC staff should review the dose modeling information provided in the DP pertaining to the licensee's proposed alternate criteria. It should use the findings and conclusions of the review under this section to evaluate the DP's compliance with 10 CFR 20.1404, "Alternate Criteria for License Termination." The staff should ensure that, at a minimum, information on the source term, exposure scenario(s), conceptual model(s), numerical analyses, and uncertainty have been included. The NRC staff may also perform a high-level review of the assumptions regarding the source term, the conceptual model of the site or building as appropriate, the exposure scenarios, the mathematical method employed, and the parameters used in the analyses and their uncertainty. The NRC staff should also review the public health and safety and protection of the environment as the basis for the alternate criteria.

## 5.5 Safety Evaluation Criteria and Review

When performing a safety evaluation, NRC staff should review the technical content of the information provided by the licensee to ensure that the licensee used defensible assumptions and models to calculate the potential dose to the average member of the critical group. The staff should also verify that the licensee provided enough information to allow an independent evaluation of the potential dose resulting from the residual radioactivity after license termination and to allow NRC staff to decide whether it has reasonable assurance that the decommissioning option will comply with license termination regulations in 10 CFR Part 20, Subpart E. A general outline and template for the development of a safety evaluation report is provided in NUREG-1757, Volume 1, Appendix G.

Specific considerations for the safety evaluation vary based on the decommissioning group associated with the site and the proposed path forward. The following subsections provide specific details regarding what should be considered based on whether NRC staff will be performing a screening review, a site-specific review resulting in the release of a site for unrestricted use, a site-specific review resulting in the release of a site for restricted use, or the release of a site using alternate criteria.

### 5.5.1 Screening Safety Review Evaluation Criteria

Evaluation considerations will vary depending on whether the submittal uses only default screening methods and parameters, or if the licensee proposes the use of site-specific parameter values. When performing a safety evaluation of a licensee's submittal proposing the use of a screening criteria NRC staff should determine if the screening criteria were used correctly by the licensee and whether the calculations provide reasonable assurance that

potential doses would not exceed dose limits. If site-specific parameters are used, the analysis is not considered a screening analysis and Sections 5.5.2 through 5.5.4 should be consulted.

When licensees use the default screening methods and parameters inherent in the DandD code by either running the computer code or using lookup tables (see Appendix H), the NRC staff will have already reviewed and accepted nearly all areas of the analysis in developing the screening tool and should only need to review the source concentrations and distribution of residual radioactivity and the overall applicability of using the screening method with the associated residual radioactivity.

Specific considerations associated with the evaluation of surface soils and building surfaces, discussed below, also need to be reviewed. If the licensee did not directly calculate the dose from residual radioactivity but instead derived, or proposed to use, lookup tables to derive DCGLs, the licensee should provide the basis or some discussion for their use. In cases where sufficient support is provided for a proposed approach, NRC staff would only need to review the information on the configuration of the residual radioactivity and the appropriate screening criteria section, below. For licensees who plan to use screening criteria, residual radioactivity should be reasonably represented by a homogeneous source (i.e., the source should not be overly heterogeneous). In general, there should be no elevated areas of residual radioactivity (e.g., "hot spots") above the screening values. DandD is limited in its capability to consider the sensitivity of dose with respect to the area of residual radioactivity. However, if one or more elevated areas or "hot spots" of residual radioactivity are expected to be present at the site following decommissioning, these areas will also need to be considered in demonstrating compliance with radiological criteria for license termination. Appendix I, Sections I.2.3 and I.3.3.3, contain additional guidance on considering elevated areas of residual radioactivity. If DCGLs are developed for the site, it is possible to consider elevated areas of residual radioactivity through use of $DCGL_{EMC}$s and address these areas during the FSS. Because it is more difficult to develop $DCGL_{EMC}$s when screening approaches are used, it is important for the licensee to engage NRC staff as early as possible on acceptable approaches for deriving these DCGLs or considering elevated areas.

When reviewing a screening analysis for building surfaces and surface soils NRC staff should use the following criteria:

- Source Configuration

  The NRC staff should confirm that the actual measurements, facility history, and planned remedial action(s) support the source configuration used in the modeling by reviewing the portions of the DP on facility history, radiological status, and planned remedial action(s).

  When evaluating building surfaces the NRC reviewer should verify both the areal extent of residual radioactivity and the depth of penetration of the residual radioactivity into the building surfaces. The NRC reviewer should also determine if the physical configuration of the residual radioactivity can adequately be assumed to be a thin layer of residual radioactivity on the building surfaces.

  Similarly, when evaluating surface soils NRC staff should review both the areal extent of residual radioactivity and the depth of penetration of the residual radioactivity into the soil. The reviewer should determine if the physical configuration of the residual

radioactivity can adequately be assumed to be a layer of surface soil containing residual radioactivity (i.e., no subsurface radioactivity).

If, during the review, it is determined that the residual radioactivity is not limited to the building surfaces or the surface soil then use of the default screening criteria is not warranted without additional justification. The NRC reviewer should reclassify the licensee as Group 4 and evaluate the modeling using Section 5.5.2.

- Residual Radioactivity Spatial Variability

The NRC staff should review the licensee's information on conditions before and those projected after the decommissioning alternative is complete. Based on this information, the NRC should determine whether it is appropriate to assume homogeneity (1) for the whole facility or (2) for subsections of the facility when evaluating building surfaces.

Similarly, for surface soils, NRC staff should review the licensee's information to determine whether it is appropriate to assume homogeneity (1) for the entire affected area or (2) for major subsections of the site. The NRC staff should then review the adequacy of the licensee's determination of a representative value (or range of values) for the residual radioactivity concentration representing the source(s). For elevated areas, the reviewer could use the general concepts related to $DCGL_{EMC}$s and the more detailed guidance on considering elevated areas of residual radioactivity discussed in Sections I.2.3 and I.3.3.3 of Appendix I.

- Conceptual Models

A detailed review of the conceptual model is not necessary as the NRC staff addressed these topics when it established the default screening methods. However, the reviewer should verify that the site and DandD's conceptual models are compatible. Situations that would not allow the use of the DandD code as a screening tool would include those where the source is not predominantly on building surfaces (i.e., volumetric source) or use of the building could lead to higher predicted doses compared to the building occupancy exposure scenario. A list of screening values for beta and gamma emitters can be found in Appendix H, Table H.1.

In the case of surface soil evaluations, situations that would not allow use of the DandD code as a screening tool would include those where the source is not predominantly present in the surface soil, residual radioactivity is in the aquifer, or sites with infiltration rates higher than the vertical saturated hydraulic conductivity (i.e., resulting in surface runoff or a bathtub effect). Alternatively, additional information could be provided to support a determination that the results tend to overestimate the dose. A list of screening values can be found in Appendix H, Table H.2.

- Execution of the DandD Computer Code Dose Calculations

If the licensee has used the DandD computer code to calculate the dose based on either current concentrations or projected final concentrations, the NRC staff should verify the following items, as applicable:

- o The residual radioactivity is limited to building surfaces or surface soil.
- o The total dose calculated includes all sources of residual radioactivity.
- o The output reports verify that no parameters (other than source concentrations) were modified.
- o The licensee has used the 90th percentile (or higher percentile) of the dose distribution to compare with the dose limit.
- o If the appropriate annual peak dose is greater than 0.025 mSv (2.5 mrem), the removable fraction of the residual radioactivity is 10 percent or less at the time of license termination, or the removable fraction has been adjusted, as explained in footnote a in Table H.1. Note: This item only applies to residual radioactivity on building surfaces.

- DCGLs from the DandD Code or Lookup Tables

The licensee may use either the DandD computer code or the published lookup tables for building surfaces in Appendix H, Table H.1, or for surface soils in Appendix H, Table H.2, to establish radionuclide-specific DCGLs equivalent to 0.25 mSv/y (25 mrem/y). If the licensee proposes to use radionuclide-specific DCGLs, the NRC staff should verify that the following conditions are true, as applicable:

(1) The residual radioactivity is limited to building surfaces or surface soil.

(2) If more than one radionuclide is involved, there is reasonable assurance that the sum of fractions (concentrations divided by DCGLs) (see Section 2.7) is no greater than 1.

(3) For building surfaces, if the residual radioactivity is greater than 10 percent of the respective screening DCGLs (Table H.1 from Appendix H of this volume), the removable fraction is 10 percent or less at license termination, or the removable fraction has been adjusted, as explained in footnote a in Table H.1.[8] Note: This item only applies to residual radioactivity on building surfaces.

If the licensee has used the DandD computer code to calculate the radionuclide-specific DCGLs, the NRC staff should also verify that the following two conditions are true:

(1) The output reports verify that no parameters (other than entering unit concentrations) were modified.

[8] The DandD default scenario assumes that only 10 percent of the building surface residual radioactivity is removable and available for resuspension. Only at greater than 10 percent of the dose limit does the assumption become important because, in the extreme case of a 100 percent removable fraction for radionuclides whose dose is dominated by the inhalation pathway, the result could only be at most 10 times higher which corresponds to the dose limit of 0.25 mSv/y (25 mrem/y).

(2) The licensee has used the 90th percentile (or higher percentile) of the dose distribution to derive the DCGLs.

- Compliance with Regulatory Criteria

The licensee's projections of compliance with regulatory criteria are acceptable provided that the NRC staff has reasonable assurance that at least one of the following is true:

(1) The final concentrations result in a peak annual dose of less than 0.25 mSv (25 mrem) and the licensee has committed to calculating the annual dose using a screening analysis at license termination.

(2) The planned DCGLs are equal to or less than those provided by the screening criteria, and the licensee has committed to ensuring the sum of fractions is no greater than 1, if applicable.

### 5.5.2 Evaluation Criteria for Decommissioning Groups 4 – 5 (Unrestricted Release Using Site-Specific Information)

The NRC staff should determine the acceptability of the licensee's projections of radiological impacts from residual radioactivity on the average member of the critical group during the compliance period. The information in the DP is acceptable if it is sufficient to ensure a defensible assessment of the possible future impacts from the residual radioactivity. The licensee's assessment can be either realistic or prudently conservative. The information should allow an independent NRC staff evaluation of the assumptions used (e.g., source configuration, applicable transport pathways) and possible doses to the average member of the critical group.

The NRC staff should review the following information, as necessary, for each dose assessment of residual radioactivity that the licensee has submitted for the various decommissioning options.

- Source Configuration and Release

The NRC staff should review the licensee's dose modeling source term assumptions and compare them with the current site information and planned remedial activities. The model should be an appropriate abstraction (or simplification) of this information. Four key areas of review for the source term assumptions are (1) source configuration, (2) residual radioactivity spatial variability, (3) release mechanisms, and (4) chemical form(s). Section I.2 from Appendix I of this volume provides additional guidance.

(1) Source Configuration

The NRC staff should confirm that the actual measurements, facility history, and planned remedial action(s) support the source configuration used in the modeling by reviewing the information in the facility history, radiological status, and planned remedial action(s) portions of the DP. The review should include both the areal extent of residual radioactivity and the depth (for soil or buildings) or volume (for groundwater or buried material) of the residual radioactivity. The NRC reviewer should also determine if the information provided supports the configuration assumptions used in the exposure scenario(s) as well as the computer and mathematical model(s) (e.g., a thin layer of residual radioactivity on the building surfaces).

(2) Residual Radioactivity Spatial Variability

The NRC staff should review the licensee's residual radioactivity concentration values for conditions both before and projected after the decommissioning alternative is complete. For this subsection, NRC staff should review the spatial extent and the degree of heterogeneity in the values. Based on this information, the NRC staff should determine whether it is reasonable to assume homogeneity for each source for either (1) the whole site or (2) specific subsections of the site. The staff should then review the adequacy of the licensee's determination of a representative value (or range of values) for the residual radioactivity concentration in the dose model. For elevated areas, the reviewer could use the general concepts related to $DCGL_{EMC}$s and more detailed guidance on considering the elevated areas of residual radioactivity discussed in Sections I.2.3 and I.3.3.3 from Appendix I of this volume.

If the licensee used dose modeling to develop DCGLs instead of estimating final concentrations and if spatial variability is a concern, then the licensee should develop $DCGL_{EMC}$s and provide this information in the DP or FSS. The NRC staff should verify that spatial variability is adequately considered in DCGL development and cleanup criteria are compatible with the assumptions made for dose modeling.

(3) Release Mechanisms

The NRC staff should review the licensee's assumptions on mechanisms controlling release of radioactivity from the source. Commonly used decommissioning dose modeling codes such as DandD and the RESRAD family of codes contain built-in release models. RESRAD-OFFSITE offers additional source release models that are not available in DandD and RESRAD-ONSITE. For additional information, see Section I.2.

(4) Chemical Form

The NRC staff should review the licensee's assumptions about the adequacy of the chemical form of the residual radioactivity and should determine whether the licensee has considered possible chemical changes that may occur during the time period of interest. Without any justification of possible chemical forms, the analysis should use the bounding chemical form(s) (e.g., the chemical form(s) that give the individual the highest dose per unit intake, as described in Federal Guidance Report Number 11 (EPA, 1988b)). The licensee should provide an acceptable rationale for other assumptions. Some acceptable rationales for using other chemical forms are (1) chemical forms that would degrade quickly in the environment (e.g., uranium hexafluoride ($UF_6$)) or (2) the unavailability of an element or conditions to realistically form that molecule (e.g., strontium titanate or high-fired uranium dioxide ($UO_2$)[9]).

- Critical Groups, Exposure Scenarios, Pathways, Identification, and Selection

In its review, the NRC staff should confirm that the licensee has identified and quantified the most significant exposure scenarios based on available site- or facility-specific information, as well as the basis and justification for the licensee's selected critical group. For exposure scenarios in which possible environmental pathways have been

[9] Strontium titanate and high-fired $UO_2$ are relatively insoluble and, therefore, these chemical forms would be expected to be retained in the pulmonary region for longer periods of time, delivering a greater dose to the lung compared to other chemical forms. However, high-fired uranium dioxide is only expected to be created under extremely high temperatures, above 800 degrees Celsius (C), and strontium titanate is considered to be artificially created, although strontium titanate has been found naturally in remote areas of the world.

modified or eliminated, the NRC staff should review the justifications provided by the licensee. Section I.3 has additional guidance on these subjects.

(1) Exposure Scenario Identification

The compliance exposure scenario is based on the location and type of source (e.g., contaminated walls), the reasonably foreseeable land use, the general characteristics and habits of the critical group (e.g., an adult light-industry worker), and the possible pathways that describe how the residual radioactivity would incur dose in humans. The licensee should provide justification for the exposure scenario(s) evaluated.

The licensee should justify the possible land use(s) the site might experience in the future and create exposure scenarios consistent with these uses. The licensee should also justify its selection of a compliance exposure scenario from the possible exposure scenarios derived from the current and projected land uses. The compliance exposure scenario should result in the greatest exposure to the average member of the critical group for all exposure scenarios given the mixture of radionuclides. A licensee may choose to make a bounding assumption for land use to derive the exposure scenario (e.g., assuming a rural land use for an urban location) or base the exposure scenario on the reasonably foreseeable land use that results in the highest dose. The other, less likely but plausible exposure scenarios are considered to better risk inform the decision.

If the compliance exposure scenario is based on the reasonably foreseeable land use, the licensee should justify the exposure scenario based on discussions with land planners, meetings with local stakeholders, trending analyses of land use for the region, or comparisons with land use in similar alternative locations. The time period of interest for possible land use changes is within 100 years, depending on the rate of change in the region and the peak exposure time. Note that the 100-year timeframe described here is only for estimating future land uses; the licensee must evaluate doses that could occur over the 1,000-year time period specified in the LTR. The licensee should also identify what land uses are less likely but plausible and evaluate exposure scenarios consistent with these less likely but plausible land uses. If use of reasonably foreseeable land use exposure scenarios results in eliminating a significant number of exposure pathways, the licensee may need to evaluate offsite exposure scenarios to ensure they do not result in greater exposures, when demonstrating compliance with the radiological criteria for license termination.

The licensee should provide a quantitative analysis or a qualitative argument discounting the need to analyze all exposure scenarios generated from the reasonably foreseeable land uses. The level of detail can vary between exposure scenarios, and the licensee is expected to use simple analyses to limit the number of detailed exposure scenarios. The licensee may use screening or generic analyses to assist in determining the critical exposure scenario for compliance. With a mixture of radionuclides, more than one compliance exposure scenario may be needed. The peak dose from the exposure scenario(s) with the highest dose should be used to demonstrate compliance.

Similarly, the licensee may provide either a quantitative analysis or a qualitative argument discounting the need to analyze all exposure scenarios considering less likely but plausible land uses. The staff will use the results of these analyses to evaluate the degree of sensitivity of dose to overall exposure scenario assumptions (and the associated parameter assumptions). The reviewer will consider both the

magnitude and time of the peak dose from these exposure scenarios. If peak doses from the less likely but plausible land use exposure scenarios are significantly above the dose standard, the licensee would need to provide greater assurance that the exposure scenario is less likely to occur, especially during the period of unacceptably high dose.

The screening exposure scenarios for building surface residual radioactivity and soil residual radioactivity are described in NUREG-1549 and NUREG/CR-5512, Volumes 1, 2, and 3. Dose evaluations that use these exposure scenarios (i.e., the licensee changes parameter values or mathematical method but does not change the general exposure scenario) are acceptable if the exposure scenario is appropriate for the situation. In DPs where the licensee eliminates certain pathways, with justification, but still maintains the same general exposure scenario category, the NRC staff should find the exposure scenario identification to be acceptable. For example, a licensee may eliminate the use of groundwater because the near surface aquifer has total dissolved solids of 30,000 milligrams per liter (mg/L). The licensee still evaluates the impacts from crops grown in the residual radioactivity, but irrigation is provided by a noncontaminated source and therefore, the screening exposure scenario of a residential farmer is maintained.

(2) Critical Group Determination

In general, critical groups exposed to multiple exposure pathways result in higher doses than groups that have more limited interaction with residual radioactivity. NUREG-1549 and the NUREG/CR-5512 series detail the critical group assumptions for the screening exposure scenarios. In DPs where the licensee has used the screening exposure scenarios, the reviewer should verify that the proposed critical group satisfies the assumptions listed in NUREG-1549 and the NUREG/CR-5512 series.

The licensee should provide either qualitative or quantitative justification that the critical group is the highest exposed group for the assumed land use(s). The selection of the critical group may be dependent on the assumption of the relative mixture of radionuclides and sources of residual radioactivity present at the site. The licensee should justify its compliance approach in these cases, as well as for the critical exposure scenario for less likely but plausible exposure scenarios.

(3) Exposure Pathways

The DP should describe the exposure pathways to which the critical group is exposed, except for cases where the licensee is using the screening exposure scenarios and critical groups without modification. If the licensee has chosen to modify the screening exposure scenario, the changes should be justified. In general, the justification should be based on physical limitations or situations that would not allow individuals to be exposed as described in the exposure scenario. The exposure pathways should therefore be consistent with the land use assumptions, exposure group behavior, and physical site conditions.

For example, acceptable justifications for removing the groundwater pathway include the following: (1) the near surface groundwater is neither potable nor allowed to be used for irrigation, (2) the aquifer volume is insufficient to provide the necessary yields, and (3) there are current (and informed consideration of future) land use patterns that would support elimination of the groundwater pathway (e.g., only short-lived radionuclides are present at the site, which is currently located in an industrial

section of an urban area with restrictions on groundwater use). Justification of water quality and quantity of the saturated zone should be based on the classification systems used by EPA or the State, as appropriate. In cases where the aquifer is classified as not being a source of drinking water but is considered adequate for stock watering and irrigation but not a viable source of drinking water, the licensee can eliminate (i.e., does not need to consider) the drinking water pathway (and the fish pathway—depending on the exposure scenario). The licensee would still maintain the irrigation and meat and milk pathways, if appropriate, for the land use assumptions.

Another example would be a rural site with a relatively small discrete outdoor area of residual radioactivity (compared to the area assumed in the default exposure scenarios). In this situation, it may be appropriate, based on the area of residual radioactivity, that gardening of some vegetables and fruits would still be an assumption, but the area would not be large enough to allow one to grow grain or raise animals for meat or milk.

- Conceptual Models

The NRC staff should review the adequacy of the conceptual model(s) used by the licensee.

The conceptual model should qualitatively describe the following:

- o the relative location and activities of the critical group
- o both the hydrologic and environmental transport processes important at the site
- o the dimensions, location, and spatial variability of the source represented in the model
- o the major assumptions made by the licensee in developing the conceptual model

The NRC reviewer should verify that the licensee adequately addressed the site conditions in the conceptual model and simplifying assumptions. Section I.4 has additional guidance on these subjects.

- Calculations and Input Parameters

In its review, the NRC staff will confirm that the licensee has used a mathematical model that is an adequate representation of the proposed conceptual model and the exposure scenario. Section I.5 contains additional guidance on this subject.

(1) Execution of DandD Computer Code

If the licensee uses the DandD computer code in its analysis, the NRC staff should verify the following points:

- o The residual radioactivity is limited to the surface (either building or near surface soil, as appropriate).
- o The total dose calculated includes all sources of residual radioactivity.
- o The site conceptual model is adequately represented by DandD's inherent conceptual model.

- For residual radioactivity present on building surfaces, the licensee has modified the resuspension factor, as necessary, to account for the removable fraction expected to be present at the time of decommissioning. The default removable fraction assumed in DandD is 10 percent. If the removable fraction is expected to be greater than 10 percent, the licensee should account for higher removable fractions that might increase the resuspension factor.[10] If site conditions are consistent with assumptions in NUREG-1720, "Re-evaluation of the Indoor Resuspension Factor for the Screening Analysis of the Building Occupancy Scenario for NRC's License Termination Rule," with respect to activities and exposure scenarios, ventilation conditions, and low removable fractions at the time of decommissioning, the NUREG-1720 recommended resuspension factor value or parameter distribution may be used with minimal justification.
- For sites eliminating pathways, the licensee has used the appropriate parameters in the DandD code as "switches" to turn off the pathways without unintentionally removing others. For example, to remove the groundwater pathways, the licensee should set the drinking water rate, irrigation rate, and pond volume parameter values to zero.
- The licensee has provided adequate support for site-specific parameter values or ranges, has not defaulted to default values, and has appropriately considered parameter correlations.
- For modifications of behavioral parameters, the changes should be based on acceptable changes in the critical group, and the mean values of the behavioral parameters should be used, although use of the ranges is also acceptable.
- If the licensee has randomly sampled the parameter ranges in DandD, it has used the "peak of the mean" or "mean of the peaks" dose to calculate the dose or derive the DCGLs.

(2) Other Mathematical Methods

If the licensee uses other mathematical methods or codes, the NRC reviewer should verify the following:

- The mathematical conceptual model is compatible with the site's conceptual model (e.g., RESRAD-ONSITE would not be an acceptable mathematical method for sites with building surface residual radioactivity).
- For each parameter or parameter set, the licensee has adequately justified the parameter value or range. For modifications of behavioral parameters, the licensee should base the changes on acceptable changes in the critical group and use the mean value (or full range) of the behavior.
- If the inhalation dose can be significant (e.g., due to the presence of alpha-emitting radionuclides such as uranium or thorium), the NRC staff should review the resuspension factor or rate and the assumptions about the degree of removable residual radioactivity.

[10] The default removable fraction is multiplied by the loose resuspension factor in DandD to derive the resuspension factor. Either the removable fraction or the resuspension factor can be adjusted to account for removable fractions greater than the default value of 10 percent.

- If the licensee is performing a probabilistic analysis, it has used the "peak of the mean" or "mean of the peaks" dose to either calculate the dose or derive the DCGLs.

- Uncertainty Analysis

The NRC staff should review the licensee's discussion of the uncertainty resulting from the physical parameter values used in the analysis. The review should focus on the uncertainty analysis for the critical pathways or parameters. Reviewers should expect that the degree of uncertainty analysis depend on the level of complexity of the modeling (e.g., generally qualitative discussions for simple modeling to quantitative analyses for more complex sites). The overall acceptability of the uncertainty analysis should be evaluated on a case-by-case basis. Section I.7 of Appendix I and Appendix Q contain additional guidance on these subjects.

If the licensee evaluated exposure scenarios based on reasonably foreseeable land uses, it should provide either a quantitative analysis of or a qualitative argument discounting the need to analyze all exposure scenarios generated from the less likely but plausible land uses. The staff will use the results of these analyses to evaluate the degree of sensitivity of the dose to overall exposure scenario assumptions (and the associated parameter assumptions). The reviewer will consider both the magnitude and time of the peak dose from these exposure scenarios. If peak doses from the less likely but plausible land use exposure scenarios are significantly above the dose standards, the licensee would need to provide greater assurance that the exposure scenarios are unlikely to occur, especially during the period of unacceptably high dose.

- Compliance with Regulatory Criteria

The licensee's projections of compliance with regulatory criteria are acceptable, provided that the NRC staff has reasonable assurance of the following:

(1) The licensee has adequately characterized the source and applied a technically defensible source term.

(2) The licensee has analyzed the appropriate exposure scenario(s) and found that the exposure group(s) adequately represents a critical group.

(3) The mathematical method and parameters used are appropriate for the exposure scenario and parameter uncertainty has been adequately addressed.

(4) For deterministic analyses, the peak annual dose to the average member of the critical group for the appropriate exposure scenario(s) for the option is less than (or equal to) 0.25 mSv (25 mrem) or was used to calculate $DCGL_W$.

(5) For probabilistic analyses, the "peak of the mean" or "mean of the peaks" dose to the average member of the critical group for the appropriate exposure scenario(s) for the option is less than (or equal to) 0.25 mSv (25 mrem) or was used to calculate $DCGL_W$.

(6) Either one of the following:

- The licensee has committed to using a specific exposure scenario, model, and set of parameters with the final survey results to show final compliance with the dose limit.

- The licensee has committed to using radionuclide-specific DCGLs and will ensure that the total dose from all radionuclides will meet the requirements of Subpart E by using the sum of fractions.

### 5.5.3 Safety Evaluation Criteria for Decommissioning Group 6 (Restricted Release)

As discussed previously, sites in Decommissioning Group 6 are being considered for restricted release using site-specific dose analyses. Specific details describing the types of sites that can be considered for Decommissioning Group 6 are included in NUREG-1757, Volume 1.

The majority of the criteria used by the NRC staff to assess the acceptability of a site for restricted release are the same as criteria used for approving Decommissioning Groups 4 – 5. This includes specific areas of consideration, including the source configuration, release mechanisms, and chemical form of the waste. Similarly, likely exposure scenarios and corresponding exposure pathways as well as the modeling approach and possible uncertainty issues are also evaluated. Specific differences that need to be considered when assessing whether the DP is in compliance with 10 CFR 20.1403 include the following:

- When assessing whether the licensee has identified and quantified the most significant exposure scenarios based on available site- and facility-specific information, dose considerations will need to be made for a minimum of two sets of exposure scenarios. One set of exposure scenarios addresses the situation when institutional controls are in place and working properly. The other set of exposure scenarios addresses the possible doses that may occur if institutional controls are assumed to no longer be in effect. For purposes of the evaluation, the licensee should assume the institutional controls fail at time = 0 years. The NRC staff should review the basis and justification for the licensee's selected critical group for each exposure scenario. Section I.3 of Appendix I and Appendix M contain additional guidance on these specific areas.

- As discussed in the previous section for Decommissioning Groups 4 – 5, evaluations of the compliance exposure scenarios proposed by the licensee includes consideration of the location and type of source (e.g., contaminated walls), the reasonably foreseeable land use, the general characteristics and habits of the critical group (e.g., an adult light-industry worker), and the possible pathways that describe how the residual radioactivity would incur dose in humans. When considering restricted releases associated with Decommissioning Group 6, potential limitations for the use of a specific compliance exposure scenario should also consider limitations based on the established institutional controls associated with restricted release. This is to ensure that if land uses other than the reasonably foreseeable land use was to occur in the future, significant exposures would not result.

- When establishing exposure scenarios, the licensee may use proposed restrictions as a basis for eliminating or changing specific exposure pathways. For example, for Decommissioning Groups 4 – 5, acceptable justifications for removing the groundwater pathway include the following: (1) the near surface groundwater is neither potable nor allowed to be used for irrigation, (2) the aquifer volume is insufficient to provide the necessary yields, and (3) there are current (and informed consideration of future) land use patterns that would support elimination of the groundwater pathway. For Decommissioning Group 6, specific site restrictions precluding groundwater use (e.g., permits, regulations, etc.) could also be justification for the removal of the groundwater

pathway from consideration. However, it is important to note that in the case that institutional controls are assumed to no longer be in effect, it may be necessary to evaluate the exposure scenarios and pathways eliminated for the case when institutional controls are assumed to be effective.

### 5.5.4 Safety Evaluation Criteria – Decommissioning Group 7 (Alternate Release Criteria)

Decommissioning Group 7 sites are evaluated using alternate criteria proposed by the licensee. An alternative release proposal in accordance with 10 CFR 20.1404 may allow a dose of up to 1.0 mSv/y (100 mrem/y) for baseline conditions with restrictions in place. However, for restricted release sites and specifically for the case where institutional controls are assumed to no longer be in effect, the dose may not exceed the values in 10 CFR 20.1403(e). Furthermore, the other provisions of 10 CFR 20.1403 must also be met.

The material to be reviewed by NRC staff should ensure that the licensee used defensible assumptions and models to establish and demonstrate compliance with the proposed alternate criteria. The staff should also verify that the licensee provided (1) enough information to allow an independent evaluation of the assessment resulting from the residual radioactivity after license termination and (2) reasonable assurance that the proposed decommissioning option complies with regulations. Each evaluation should be performed on a case-by-case basis.

## 5.6 Summary Review Criteria

A summary of the review criteria listed in this chapter is provided in Table 5.2 and Table 5.3. Table 5.2 provides information on review criteria for residual radioactivity found in soils. Table 5.3 provides information on review criteria for residual radioactivity associated with buildings. Both Tables 5.2 and 5.3 provide information for screening (column 1) versus site-specific reviews (column 2); and unrestricted (top) versus restricted (bottom) release scenarios. Staff should review the information provided by the licensee to ensure an adequate basis is provided to support the modeling assumptions and parameters selected commensurate with their risk-significance as determined through sensitivity and uncertainty analysis. Data of sufficient quality should be collected to ensure the technical defensibility of the modeling results and DCGLs. The DQO process should be used to guide data collection and analysis. Staff should review the DQOs, including QA/QC requirements, to ensure DQOs are met during the DCGL development and the FSS used to demonstrate compliance with release criteria.

**Table 5.2 Dose Modeling Review Criteria for Residual Radioactivity in Soils**

| Screening (Soils) | Site-Specific (Soils) |
|---|---|
| *Unrestricted Release* | |
| Review source assumptions<br>• Nuclide(s) of interest<br>• Only surface soil residual radioactivity is present (residual radioactivity is present in approximately the top 30 cm of soil)<br>• Extent of heterogeneity of residual radioactivity (i.e., survey units should be relatively homogeneous)<br>• Method of determining average concentration | Review modeling assumptions related to the source term<br>• Nuclide(s) of interest<br>• Area of residual radioactivity<br>• Thickness of residual radioactivity<br>• Extent of heterogeneity of residual radioactivity<br>• Method of determining average concentration<br>• Treatment of elevated areas (and development of $DCGL_{EMC}$s), as appropriate<br>• Chemical form of the radionuclide (to ensure dose conversion factors do not underestimate dose; bounding values could be used if information on chemical form is lacking) |
| Review exposure scenario assumptions<br>• Evaluate whether default critical group (resident farmer) is appropriate for the site and whether there are any other critical groups which could incur higher doses. | Review exposure scenario assumptions<br>• Assess whether the exposure scenario(s) and critical group(s) used to demonstrate compliance are appropriate for the site.<br>• Assess whether reasonably foreseeable exposure scenarios are considered based on land use and other data, as well as stakeholder input.<br>• Assess the adequacy of information provided to support elimination of or reduction in dose associated with various pathways of exposure.<br>• Ensure that less likely but plausible scenarios are identified and considered, as appropriate.<br>• If subsurface residual radioactivity is present, whether DCGLs are derived for multiple layers and whether excavation scenarios which would bring the material to the surface are considered. |
| Review conceptual site model assumptions<br>• Check to make sure site conditions are consistent with the built-in conceptual model in the DandD screening code (see Appendix I, Table I.5)<br>o groundwater initially free of residual radioactivity<br>o surface water sediments initially free of residual radioactivity | Review conceptual site model assumptions<br>• Check to make sure site conditions are consistent with the built-in conceptual model in RESRAD (see Appendix I, Tables I.6 and I.7)<br>• Evaluate the relative location and activities of the critical group.<br>• Evaluate the distribution of residual radioactivity in the environment (is residual |

**Table 5.2 Dose Modeling Review Criteria for Residual Radioactivity in Soils (cont.)**

| | |
|---|---|
| o infiltration rate is greater than the vertical saturated hydraulic conductivity | radioactivity initially present in the subsurface or in groundwater?)<br>• Evaluate the hydrogeological conceptual model developed for the site (e.g., depth to groundwater, subsurface layers or materials, aquifer thickness).<br>• Evaluate important environmental transport processes. |
| Review screening method<br>• Evaluate whether screening or look-up values (see Appendix H) are being used.<br>• If screening values are being used and multiple radionuclides are present, check to make sure that the sum of fractions value (concentrations divided by screening values) is not greater than 1.<br>• The final concentration should result in a peak annual dose of less than 0.25 mSv (25 mrem/) | Review mathematical models<br>• Evaluate the mathematical models used to assess dose (guidance in this volume focuses on use of DandD and probabilistic RESRAD, which were sponsored by the NRC for use in decommissioning) and compatibility with the site conceptual model.<br>• If custom models or non-traditional codes are used, evaluate whether the mathematical models are appropriate and compatible with the conceptual site model and have proper QA/QC (see Appendix I, Section I.5). |
| Review Parameter Assumptions<br>• Ensure that the default parameters developed for DandD (see NUREG/CR-5512, Volume 3) are used.<br>• The only parameters that require user input are the list of radionuclides and their associated concentrations. | Review Parameter Assumptions<br>• Review appropriateness of physical parameters developed for the site (licensees may use DandD default behavioral and metabolic parameters)<br>• If changes are made to DandD default behavioral and metabolic parameters, ensure that the changes are well supported for the exposure scenario and average member of the critical group.<br>• Evaluate whether sensitivity and/or uncertainty analysis are performed to identify the most risk-significant parameters affecting dose.<br>• Ensure that an adequate level of support is provided for the most risk-significant parameters. |
| Consideration of Uncertainty<br>• If modeling is performed to develop screening values, ensure that DandD and the 90$^{th}$ percentile of the dose distribution is used. | Consideration of Uncertainty<br>• Evaluate parameter ranges to ensure that parameter uncertainty is appropriately considered in the dose modeling. Default parameter distributions are available in RESRAD and can be used to assist with performance of probabilistic sensitivity analysis.<br>• If a deterministic analysis is used, ensure that the values selected for risk-significant parameters are sufficiently conservative. |

**Table 5.2 Dose Modeling Review Criteria for Residual Radioactivity in Soils (cont.)**

| | |
|---|---|
| | • If a probabilistic approach is used, ensure that the parameter ranges are adequately supported particularly for risk-significant parameters and that overly broad distributions do not lead to risk dilution for parameters that primarily affect the timing of peak dose.<br>• Evaluate the dose metric used (peak of the mean or mean of the peak)<br>• The final concentrations should result in a peak annual dose of less than 0.25 mSv (25 mrem/) |
| *Restricted Release* | |
| Not applicable | All of the above review criteria apply for restricted release. Additional review criteria are listed below. |
| | Evaluate two sets of exposure scenarios for the case when<br>• Institutional controls are in effect<br>• Institutional controls are no longer in effect (at time=0 years) |
| | Evaluate whether the licensee has appropriately considered off-site receptors and identified the critical group in the case where restrictions are in place. |
| | In the case of loss of institutional controls, evaluate whether the licensee considers degradation of engineered barriers and only the passive performance of the barriers assuming no active maintenance. |
| | Evaluate the use of dose modeling information to support ALARA evaluations. |

**Figure 5.3 Dose Modeling Review Criteria for Residual Radioactivity in Buildings**

| Screening (Building) | Site-Specific (Building) |
|---|---|
| *Unrestricted Release* | |
| Review source assumptions<br>• Residual radioactivity is present on building surfaces (no volumetrically contaminated building materials)<br>• Extent of heterogeneity of residual radioactivity<br>• Method of determining average concentration | Review source assumptions<br>• Nuclide(s) of interest<br>• Area of residual radioactivity<br>• Thickness of residual radioactivity (thin surface or volumetric building contamination)<br>• Extent of heterogeneity of residual radioactivity<br>• Method of determining average concentration<br>• Treatment of elevated areas (and development of $DCGL_{EMC}$s), as appropriate<br>• Chemical form of the radionuclide (to ensure dose conversion factors do not underestimate dose; bounding values could be used if information on chemical form is lacking) |
| Review exposure scenario assumptions<br>• Evaluate whether default critical group (building occupancy) is appropriate for the site and whether there any other critical groups which could incur higher doses. | Review exposure scenario assumptions<br>• Assess whether the exposure scenario(s) and critical group(s) used to demonstrate compliance are appropriate for site conditions.<br>• If volumetric contamination is present, an evaluation of whether the building renovation scenario is more limiting and should be considered. |
| Review conceptual site model assumptions<br>• Check to make sure site conditions are consistent with the conceptual model built-in the DandD screening code<br>o only building surface residual radioactivity is present and<br>o loose contamination is 10 percent or less or the removable fraction has been adjusted | Review conceptual site model assumptions<br>• Evaluate the relative location and activities of the critical group.<br>• Evaluate the distribution of residual radioactivity in the building (surface versus volumetric; floors, walls, ceiling, equipment, piping, or sewer lines).<br>• Evaluate representation of the building in the model (geometry and building ventilation of rooms, building materials). |
| Review mathematical models and screening method<br>• Evaluate whether screening or look-up values (see Appendix H) are being used.<br>• If screening values are being used and multiple radionuclides are present, check to make sure that the sum of fractions (concentrations divided by screening values) is no greater than 1. | Review mathematical models<br>• Evaluate the mathematical models used to assess dose (guidance in this volume focuses on use of DandD and RESRAD-BUILD, which were sponsored by the NRC for use in decommissioning) and compatibility with the site conceptual model. |

**Figure 5.3 Dose Modeling Review Criteria for Residual Radioactivity in Buildings (cont.)**

| | |
|---|---|
| • If modeling is performed to develop screening values, ensure that DandD is used. | • If custom models or non-traditional codes are used, evaluate whether the mathematical models are appropriate and compatible with the conceptual site model and have proper QA/QC (see Appendix I, Section I.5). |
| Review Parameter Assumptions<br>• Ensure that the default parameters developed for DandD (see NUREG/CR-5512, Volume 3) are used.<br>• The only parameters that require user input are the list of radionuclides and their associated concentrations. | Review Parameter Assumptions<br>• Review appropriateness of parameters developed for the site (licensees may use DandD default behavioral and metabolic parameters).<br>• Evaluate whether sensitivity and/or uncertainty analysis are performed to identify the most risk-significant parameters affecting dose.<br>• Ensure that an adequate level of support is provided for the most risk-significant parameters. |
| Consideration of Uncertainty<br>• If modeling is performed to develop screening values, ensure that DandD and the 90$^{th}$ percentile of the dose distribution is used. | Consideration of Uncertainty<br>• Evaluate parameter ranges to ensure that parameter uncertainty is appropriately considered in the dose modeling. Default parameter distributions are available in RESRAD BUILD and can be used to assist with performance of probabilistic sensitivity analysis.<br>• If a deterministic analysis is used, ensure that the values selected for risk-significant parameters are sufficiently conservative.<br>• If a probabilistic approach is used, ensure that the parameter ranges are adequately supported.<br>• Evaluate the dose metric used (peak of the mean or mean of the peak) |
| *Restricted Release* | |
| Not applicable | All of the above review criteria apply for restricted release. Additional review criteria are listed below. |
| | Evaluate two sets of exposure scenarios for the case when<br>• Institutional controls are in effect<br>• Institutional controls are no longer in effect (at time=0 years) |

## 5.7 Additional Regulatory Guidance

In addition to the information provided above, there are numerous other references that the licensee or the NRC reviewer can refer to when developing and reviewing the dose modeling portions of DPs. Some of these references are listed below.

### 5.7.1 Regulatory Guidance Related to Decommissioning Groups 1 – 3 (Unrestricted Release Using Screening Criteria)

- Appendix H to this NUREG report
- NUREG-1757, Vol. 1, "Consolidated NMSS Decommissioning Guidance: Decommissioning Process for Materials Licensees"

### 5.7.2 Regulatory Guidance Related to Decommissioning Groups 4 – 5 (Unrestricted Release Using Site-Specific Information)

- Appendix I to this NUREG report
- NUREG-1549, "Decision Methods for Dose Assessment to Comply with Radiological Criteria for License Termination"
- NUREG/CR-5512, Volume 1, "Residual Radioactive Contamination from Decommissioning: Technical Basis for Translating Contamination Levels to Annual Total Effective Dose Equivalent"
- NUREG/CR-5512, Volume 2, "Residual Radioactive Contamination from Decommissioning: User's Manual, DandD Version 2.1"
- Draft NUREG/CR-5512, Volume 3, "Residual Radioactive Contamination from Decommissioning: Parameter Analysis"
- EPA, Federal Guidance Report Number 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," September 1988
- EPA, Federal Guidance Report Number 12, "External Exposure to Radionuclides in Air, Water, and Soil," September 1993

### 5.7.3 Regulatory Guidance Related to Decommissioning Group 6 (Restricted Release)

- NUREG-1549, "Decision Methods for Dose Assessment to Comply with Radiological Criteria for License Termination"
- NUREG-1573, "A Performance Assessment Method for Low-Level Waste Disposal Facilities: Recommendations of NRC's Performance Assessment Working Group"
- Draft NUREG-2175, "Guidance for Conducting Technical Analyses for 10 CFR Part 61"

# 6 ALARA ANALYSES

This chapter is applicable to Decommissioning Groups 2–7. Licensees in Decommissioning Groups 2 and 3 may only have to refer to the discussion of good housekeeping practices in Section 6.3.

## 6.1 Safety Evaluation Review Procedures

### 6.1.1 Areas of Review

The NRC staff should review the information supplied by the licensee or responsible party to determine if the licensee has developed a DP that ensures that doses to the average member of the critical group are ALARA. Information submitted should include (1) a cost-benefit analysis (or qualitative arguments) demonstrating that the applicable ALARA requirement(s) for the licensee's preferred decommissioning option will be met and (2) a description of the licensee's preferred method for showing compliance with the ALARA requirement at the time of decommissioning. If the licensee proposes to use engineered barriers or intentional mixing to meet the LTR criteria for unrestricted use, it should complete an appropriate ALARA analysis, as described in Section 3.5 of this volume (for engineered barriers) or Section 15.13 of Volume 1 (for intentional mixing). Additionally, an ALARA evaluation for restricted use should follow guidance described in Appendix N of this NUREG-1757, Volume 2. The licensee should also follow the guidance in Appendix N of this volume when evaluating the eligibility tests for restricted use (10 CFR 20.1403(a), ALARA test, or test of net public or environmental harm) and use of a higher dose limit for restricted use with institutional controls not in place (10 CFR 20.1403(e)(2)(i), test for prohibitively expensive, or test for net public or environmental harm).

### 6.1.2 Review Procedures

#### *6.1.2.1 Acceptance Review*

The NRC staff should review the DP to ensure that, at a minimum, it contains the information summarized under the above "Areas of Review." The NRC staff should review the ALARA portion of the DP without assessing the technical accuracy or completeness of the information contained therein, which it should determine during the detailed technical review. The NRC staff should review the DP table of contents and the individual descriptions under "Areas of Review" (1) to ensure that the licensee or responsible party has included this information in the DP and (2) determine if the level of detail of the information appears to be adequate for a detailed technical review.

#### *6.1.2.2 Safety Evaluation*

The material supporting the ALARA portion of the DP to be reviewed is technical in nature and specific detailed technical analysis may be necessary. The NRC staff should evaluate the licensee's dose estimates for various alternatives using the appropriate guidance in Chapter 5 of this volume and should evaluate the licensee's cost estimates using the guidance in Section 4.1 from NUREG-1757, Volume 3.

## 6.2 Acceptance Criteria

### 6.2.1 Regulatory Requirements

10 CFR 20.1101(b), 20.1402, 20.1403(a), 20.1403(b), 20.1403(e), and 20.1404(a)(3)

### 6.2.2 Regulatory Guidance

Appendix N of Volume 2 of this NUREG report

### 6.2.3 Information to be Submitted

The information supplied by the licensee should be sufficient to allow the NRC staff to fully understand the basis for the licensee's conclusion that projected dose or residual radioactivity concentrations and quantity (hereafter, the decommissioning goal) are ALARA. The decommissioning goal should be established at the point where the incremental benefits equal the incremental costs. The NRC staff review should verify that the following information is included in the description of the development of the decommissioning goal:

- a description of how the licensee will achieve a decommissioning goal that meets the dose limit and ALARA requirement
- a quantitative cost-benefit analysis
- a description of how costs were estimated
- a demonstration that the doses to the average member of the critical group are ALARA

The information to be submitted is also included as part of the master DP Checklist provided in this NUREG report (see Section VII from Appendix D of Volume 1).

## 6.3 Evaluation Criteria

### 6.3.1 Evaluation of Good Practice Efforts

For ALARA during decommissioning, all licensees should use typical good practice, such as floor and wall washing, removal of readily removable radioactivity in buildings or in soil areas, and other good housekeeping practices. In addition, the FSSR should describe how the licensee employed these practices to achieve the final activity levels.

### 6.3.2 Evaluation of Cost-Benefit ALARA Analyses

The NRC staff review should verify that the qualitative descriptions provide reasonable assurance that the activities and decommissioning goal should result in doses to the average member of the critical group that are ALARA. For those situations in which a licensee prepares cost-benefit analyses, the NRC staff should ensure that the analyses are developed using the methodology described in Appendix N and are applied as described in the following text.

### 6.3.3 When Cost-Benefit Analyses are Unnecessary

In the following circumstances, the results of an ALARA analysis are known on a generic basis and the NRC staff considers that an analysis is not necessary (see Appendix N of this volume for more details):

- unrestricted use where excavated soil would be shipped to a LLW disposal facility for disposal
- soil removal at a site to meet the unrestricted use dose criteria of 25 mrem/y
- remediation of building surfaces or surface soil to the NRC default screening levels (see Appendix H for information about the screening levels)
- that no residual radioactivity distinguishable from background will remain at the site at termination
- that loose residual radioactivity on building surfaces has been or will be removed

### 6.3.4 Calculation of Benefits

Appendix N of this volume discusses five different possible benefits: (1) collective dose averted, (2) regulatory costs avoided, (3) changes in land values, (4) esthetics, and (5) reduction in public opposition. Numerical estimates will generally only be available for the first three benefits, if they are appropriate. The licensee can make a qualitative analysis of the benefits, especially if the costs are large (e.g., no matter what the change in land value is, the costs will exceed the benefits). In most comparisons between alternatives in the same class (e.g., both alternatives result in unrestricted release), the only important benefit should be the collective dose averted. In comparisons between restricted and unrestricted release, the other benefits can become important.

The collective dose averted is generalized as the incremental dose difference between the licensee's approach (hereafter, preferred option) and the alternative under analysis. Therefore, the NRC staff should ensure that the licensee has calculated the benefits correctly by using the correct population density, area, and averted dose. This may require a technical analysis of the dose modeling, and the reviewer should use Chapter 5 for these cases. If the licensee has used discounting, the NRC staff should ensure that the proper rates were used. The licensee is not required to discount because the discount reduces the benefits of averting dose in later time periods.

One acceptable method of compliance with 10 CFR 20.1403(a) is to demonstrate that cleanup to the unrestricted release criteria is beyond ALARA considerations. In this case, a beneficial estimate should include costs that would be avoided if the site were to be released for unrestricted use, such as site control and maintenance costs, as well as an estimate of the additional regulatory costs associated with termination of a restricted site (e.g., development of an environmental impact statement, public meetings). Appendix N of this volume contains more details on compliance with ALARA requirements for restricted use.

The NRC staff should ensure that the licensee has properly documented the basis for any estimates of changes in land values. Acceptable sources of such estimates include

governmental assessors (e.g., county, State) or real estate agents familiar with the local area and the issues involved.

### 6.3.5 Calculation of Costs

The NRC staff should verify that the licensee has adequately estimated the effective monetary costs of the incremental remediation by using the equations in Appendix N of this volume. To review the calculated monetary costs of the incremental remediation, the NRC staff should use Section 4.1 of NUREG-1757, Volume 3, with the following changes (this may require calculating total cost estimates for the preferred option and each alternative):

- The cost estimate should be based on actual costs expected to be incurred by decommissioning the facility and should not assume that the work will be performed by an independent third-party contractor.

- The cost estimate *does* take credit for (1) any salvage value that might be realized from the sale of potential assets during or after decommissioning or (2) any tax reduction that might result from paying decommissioning costs or site control and maintenance costs.

- The decommissioning cost estimates should reflect the actual situation rather than maximized assumptions.

For each of the cost terms (e.g., disposal costs, worker fatalities), the incremental difference between the preferred and the alternative options may be negative (i.e., the alternative may cost less than the preferred option).

### 6.3.6 Compliance Methods at the Time of Decommissioning

There are two approaches to demonstrate compliance with the ALARA requirement at the end of decommissioning: (1) a predetermined acceptable dose limit or concentration guideline(s) or (2) an acceptable preferred option and decommissioning goal with organizational oversight and review during decommissioning. Both options have advantages and disadvantages. The licensee establishes the compliance method, with the staff reviewing the applicability, given the site-specific information.

#### *6.3.6.1 Predetermined Compliance Measure*

Under the predetermined compliance measure, the licensee would agree to meet the dose calculated for the preferred option or the radiological concentrations associated with that dose. This could be met by either establishing deterministic concentration limits for the site or agreeing to use a specified dose scenario with associated parameters and assumptions. If the licensee's final survey results meet the self-imposed concentration limits (or dose limit), the licensee has met the ALARA requirement.

#### *6.3.6.2 Performance-Based Compliance*

Performance-based compliance allows a licensee to adjust its ALARA assessment during decommissioning to deal with actual site conditions experienced and actual costs incurred. The philosophy behind this compliance measure is very similar to how ALARA is handled during routine operations. The licensee's DP needs to meet all of the following criteria to use this approach:

- The preferred option, based on valid assumptions, would result in reducing residual activity to ALARA levels, as described above.

- The licensee has established decommissioning guidelines (either dose or concentrations) based on the DP's analysis.

- The licensee has a documented method to review the effectiveness of the remediation activities. This method should include all of the following:

  - an ALARA committee or radiation safety officer for small licensees, similar to operations requirements

  - an appropriate review frequency established

  - an acceptable set of criteria on the scope of activities or commitments that the ALARA committee can change

  - a commitment to prepare acceptable documentation of ALARA findings that result in the licensee changing its remediation activities or decommissioning guidelines

  - a commitment to provide the NRC annually with all necessary page changes to the DP because of ALARA findings

At the end of remediation, a licensee using the performance-based approach should meet the following criteria:

- The final survey results satisfy the appropriate dose limit(s).

- Any substantial weaknesses in the ALARA program that were found during licensee audits or NRC inspections have been resolved.

- Any deviation from the decommissioning goal presented in the DP is properly justified by the ALARA committee findings.

The NRC license reviewer or inspection staff should review long-term projects annually.

# 7 BIBLIOGRAPHY AND SUPERSEDED DOCUMENTS

This section provides the reference list for the main body of this volume, categorized in the following subsections by type of reference document. Chapter 4 of Volume 1 of this NUREG series provided a more general list of decommissioning references, which included statutes, decommissioning regulations, decommissioning inspection manual chapters, and decommissioning inspection procedures.

**Use of References Cited in this Volume**

This volume refers to a number of other documents for guidance. In some cases, this volume will state that the referenced guidance is approved by the NRC staff. However, in some cases, the documents are only referenced for information, and, if so, the licensee should contact the NRC staff to determine the specific applicability to a facility, as appropriate.

## 7.1 NRC Decommissioning Documents Referenced in the Main Body of Vol. 2


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— — — — —. "Working Draft Regulatory Guide on Release Criteria for Decommissioning: NRC Staff's Draft for Comment," NUREG-1500. NRC: Washington, DC. 1994d.

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— — — — —. "Residual Radioactive Contamination from Decommissioning, User's Manual, DandD Version 2.1," NUREG/CR-5512, Vol. 2. NRC: Washington, DC. 2001a.

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— — — — —. "Re-Evaluation of the Indoor Resuspension Factor for the Screening Analysis of the Building Occupancy Scenario for NRC's License Termiantion Rule—Draft Report for Comment," NUREG-1720. NRC: Washington, DC. 2002b.

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— — — — —. "Results of the License Termination Rule Analysis of the Use of Intentional Mixing of Contaminated Soil," SECY-04-0035. NRC: Washington, DC. 2004a.

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[1] Corrected pages for MARSSIM, Revision 1 (August 2000) with the June 2001 updates, are available at the NRC website: http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1575/r1/.

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## 7.2 Other NRC Documents Referenced in this Volume

*Code of Federal Regulations (CFR)*. 10 CFR Part 20, “Standards for Protection against Radiation.”

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## 7.3 Other References in this Volume

American National Standards Institute. "Characterization in Support of Decommissioning Using the Data Quality Objectives Process," ANSI N13.59:2008. ANSI: New York, NY. 2008.

Atomic Energy Act of 1954, as amended. 42 U.S.C. §§2011 et seq.

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## 7.4 Documents Superseded by this Volume

This volume supersedes the guidance documents listed in Table 7.4, and the superseded documents should no longer be used.

**Table 7.1 Documents Superseded by this Report**

| Document | Title | Date |
|---|---|---|
| NRC Memorandum | Draft Staff Guidance for Dose Modeling of Proposed Partial Site Releases | 09/2001 |
| NUREG-1727 | NMSS Decommissioning Standard Review Plan | 09/2000 |
| NUREG/BR-0241 | NMSS Handbook for Decommissioning Fuel Cycle and Materials Licensees | 03/1997 |
| Branch Technical Position | Draft Branch Technical Position: Screening Methodology for Assessing Prior Land Burials of Radioactive Waste Authorized Under Former 10 CFR 20.304 and 20.302 | 10/1996 |
| Branch Technical Position | Draft Branch Technical Position on Site Characterization for Decommissioning | 11/1994 |
| NUREG-1500 | Working Draft Regulatory Guide on Release Criteria for Decommissioning: NRC Staff's Draft for Comment | 08/1994 |
| NUREG/CR-5849 | Manual for Conducting Radiological Surveys in Support of License Termination | 06/1992 |
| Branch Technical Position | Disposal of Onsite Storage of Thorium or Uranium Wastes from Past Operations | 10/1981 |

This Volume 2 of this NUREG report also incorporates and updates numerous portions of NUREG-1727, "NMSS Decommissioning Standard Review Plan" (SRP), issued September 2000, specifically Chapters 5, 7, and 14; and Appendices C, D, and E of NUREG-1727 on dose modeling, ALARA, and facility radiation surveys. The NUREG-1727 chapters and appendices that have been incorporated into this NUREG are superseded. This three-volume NUREG series (i.e., NUREG-1757, Volumes 1, 2, and 3) supersedes both NUREG/BR-0241, "NMSS Handbook for Decommissioning Fuel Cycle and Materials Licensees," issued March 1997, and NUREG-1727 in their entirety.

# APPENDIX A

# IMPLEMENTING THE MARSSIM APPROACH FOR CONDUCTING FINAL RADIOLOGICAL SURVEYS

## A.1 Introduction

This appendix is applicable to Decommissioning Groups 2–7.

Regulations of the NRC in Title 10 of the *Code of Federal Regulations* (10 CFR) 20.1501(a) require licensees to make or cause to be made surveys that may be necessary for the licensee to comply with the regulations in 10 CFR Part 20, "Standards for Protection against Radiation."

The FSS is the radiation survey performed after an area has been fully characterized, remediation has been completed, and the licensee believes that the area is ready to be released. The purpose of the FSS is to demonstrate that the area meets the radiological criteria for license termination. The FSS is not conducted for the purpose of locating residual radioactivity; the HSA and the characterization survey perform that function.

The NRC endorses the FSS method described in NUREG-1575, "Multi-Agency Radiological Survey and Site Investigation Manual" (MARSSIM), Revision 1, issued August 2000, and references to MARSSIM sections within this appendix are specifically referring to MARSSIM Revision 1 (NRC, 2000).[1] This appendix (1) provides an overview of the MARSSIM approach for conducting a final radiological survey, (2) includes additional specific guidance on acceptable values for use in the MARSSIM method, (3) explains how to use the MARSSIM method in a way that is consistent with the dose modeling, (4) describes how to use the MARSSIM method to meet the NRC's regulations, and (5) demonstrates how to extend or supplement the MARSSIM method to certain complex situations that may be encountered, such as how to address subsurface residual radioactivity. Note that the guidance in this appendix does not replace MARSSIM, and licensees and reviewers should refer to, and use, MARSSIM for designing final radiological surveys to support decommissioning. This guidance assumes a working knowledge of the MARSSIM approach and terminology and does not attempt to provide a comprehensive overview of the entire MARSSIM. In addition, for Decommissioning Groups 1–3, licensees may also use the alternative, simpler final survey methods described in Appendix B of this volume.

Chapter 5 of MARSSIM contains survey checklists. These checklists are useful in implementing the steps of the RSSI process (Decommissioning Groups 3–7). These checklists are a useful tool for visualizing the sequential steps (i.e., design, performance, and evaluation) of the survey process. Furthermore, the use of these checklists should ensure that the necessary information is collected for each type of survey. Sites not using the RSSI process, such as Decommissioning Groups 1 and 2, should also find these checklists or parts of these checklists useful.

## A.2 Classification of Areas by Residual Radioactivity Levels

The licensee should classify site areas based on levels of residual radioactivity from licensed activities. The area classification method contained in Section 4.4 of MARSSIM is acceptable to the NRC staff. Its essential features are described below.

The licensee should first classify site areas as impacted or nonimpacted. *Impacted areas* are areas that may have residual radioactivity from the licensed activities. *Nonimpacted areas* are

[1] As of the publication of this guidance document MARSSIM, Revision 1, is the current version. MARSSIM, Revision 2, is being developed but is not reflected in this guidance. The decommissioning website should be consulted for issuance of technical reports providing guidance and a listing of any lessons learned between guidance revisions.

areas without residual radioactivity from licensed activities. Impacted areas should be identified by using knowledge of past site operations together with site characterization surveys. In the FSS, radiation surveys do not need to be conducted in nonimpacted areas. The licensee should classify impacted areas into one of the three classes, listed below, based on levels of residual radioactivity.

(1) **Class 1 Areas**: Class 1 areas are impacted areas that are expected to have concentrations of residual radioactivity that exceed the derived concentration guideline level or $DCGL_W$ (average concentrations over a wide area).

(2) **Class 2 Areas**: Class 2 areas are impacted areas that are not likely to have concentrations of residual radioactivity that exceed the $DCGL_W$.

(3) **Class 3 Areas**: Class 3 areas are impacted areas that have a low probability of containing residual radioactivity.

Surveys conducted during operations or during characterization at the start of decommissioning are the basis for classifying areas. If the available information is not sufficient to designate an area as a particular class, the area either should be classified as Class 1 or should be further characterized. Areas that are considered to be on the borderline between classes should receive the more restrictive classification.

The NRC staff recognizes that a licensee may need to reclassify Class 1 areas to Class 2, when insufficient information is available for the initial classification. If more information becomes available to indicate that another classification is more appropriate, the guidance in MARSSIM allows for classifications to be changed at any time before the FSS. For more guidance on criteria for downgrading classifications (e.g., from Class 1 to Class 2), a licensee should refer to MARSSIM, Revision 1; in particular, Sections 2.2, 2.5.2, and 5.5.3. If a licensee plans to make use of reclassification during the RSSI process, it should include in the DP the criteria and methodology it plans to use for reclassification. In addition, a licensee contemplating the use of reclassification is encouraged to contact NRC staff.

For Class 1 and 2 soils, the licensee should determine whether a significant amount of subsurface residual radioactivity is present, based on the HSA and site characterization. In this context “significant amounts of subsurface residual radioactivity” would be defined as an amount of radioactivity, or contaminated material (such as soil), that could contribute at least 10 percent of the dose criteria (see Section 3.3).

The presence of subsurface soils is important, because subsurface soils cannot typically be measured by scan instrumentation and, therefore, scan surveys are not expected to be adequate for the purposes of an FSS. Although surface soils have been associated with the top 15 centimeters (cm) of soil, which can typically be measured by scan instrumentation, the exact depth of residual radioactivity for which scan survey instrumentation can adequately detect residual radioactivity varies, based on several factors (e.g., survey instrument, radionuclide, and soil characteristics).

Determining whether there is a significant amount of subsurface residual radioactivity should not require a complex set of characterization measurements. In most cases, there will be either significant amounts of residual radioactivity or only traces (such as in occasional small pockets or from leaching from surface layers by rainwater). When there is an insignificant amount of subsurface residual radioactivity, the MARSSIM survey methods for surface measurements are

acceptable. When there is a significant amount of subsurface residual radioactivity, the licensee should modify the dose modeling and survey methods to account for it.

The HSA usually determines the presence of subsurface residual radioactivity (see Chapter 3 of MARSSIM), applying knowledge of how the residual radioactivity was deposited. Characterization surveys to detect subsurface residual radioactivity in soil are not routinely conducted, unless there is reason to expect that subsurface residual radioactivity may be present. The need to survey or sample subsurface soil will depend, in large part, on the quality of the information used to develop the HSA, the environmental conditions at the site, the types and forms (chemical and radiological) of the radioactive material used at the site, the authorized activities and the manner in which licensed material was managed during operations.

> The NRC staff's experience has shown that submittal of the DP should occur only after sufficient site characterization has occurred. The staff suggests that the DP provide sufficient information demonstrating the characterization of the radiological conditions of site structures, facilities, surface and subsurface soils, and groundwater. The NRC staff has observed that some DPs have been submitted with incomplete or inadequate characterizations of radiological conditions. A review of such DPs has shown that the lack of information makes it difficult to evaluate the rationale for the proposed classification of survey units. The NRC staff suggests that the following issues related to the use of characterization survey results and classification of survey units be considered when developing a DP:
>
> - use of operational, post-shutdown scoping, or turnover surveys as characterization surveys
> - reclassification of survey units
> - completeness of characterization survey design and results
>
> Regulatory Issue Summary 2002-02, "Lessons Learned Related to Recently Submitted Decommissioning Plans and License Termination Plans," issued January 2002, provides a detailed discussion of this issue.

## A.3 Selection and Size of Survey Units

The licensee should divide the impacted area into survey units based on the classification described above. A survey unit is a portion of a building or site that is surveyed, evaluated, and released as a single unit. The licensee should give the entire survey unit the same area classification. Section 4.6 of MARSSIM contains a method acceptable to the NRC staff for dividing impacted areas into survey units. The important features of this method are summarized here.

For buildings, it is normally appropriate to designate each separate room as either one or two survey units (e.g., floors with the lower half of walls and upper half of walls with ceiling), based on the pattern of potential of residual radioactivity. It is generally not appropriate to divide rooms of normal size (100 square meters ($m^2$) or less) into more than two survey units, because the dose modeling is based on the room being considered as a single unit. However, very large spaces such as warehouses may be divided into multiple survey units.

For soil, survey units should be areas with similar operational history or similar potential for residual radioactivity to the extent practical. Survey units should be formed from areas with the same classification to the extent practical, but if areas with more than one class are combined into one survey unit, the entire survey unit should be given the more restrictive classification. Survey units should have relatively compact shapes and should not have highly irregular (gerrymandered) shapes unless the unusual shape is appropriate for the site operational history or the site topography.

Table A.1 contains suggested survey unit areas from MARSSIM. These areas are suggested in MARSSIM, because they give a reasonable sampling density and are consistent with most commonly used dose modeling codes. However, the size and shape of a particular survey unit may be adjusted to conform to the existing features of the particular site area.

**Table A.1 Suggested Survey Unit Areas** *(MARSSIM, Rev. 1, Roadmap, Table 1 [NRC, 2000])*

| | Suggested Survey Unit Area | |
|---|---|---|
| Class | Structures | Land |
| 1 | up to 100 $m^2$ | up to 2,000 $m^2$ |
| 2 | 100 to 1,000 $m^2$ | 2,000 to 10,000 $m^2$ |
| 3 | no limit | no limit |

## A.4 Selection of Background Reference Areas and Materials

### A.4.1 Need for Background Reference Areas

Background reference areas are not needed when radionuclide-specific measurements will be used for concentrations of a radionuclide that is not present in background. Background reference areas are needed for the MARSSIM method if (1) the residual radioactivity contains a radionuclide that occurs in background, or (2) the sample measurements to be made are not radionuclide specific. However, a licensee may find it cost beneficial to consider the background for a particular radionuclide as zero or some other appropriately low value approved by the staff, recognizing that this is a risk-informed approach. The survey unit itself may serve as the reference area when a surrogate radionuclide in the survey unit can be used to determine background. For example, it may be possible to use radium (Ra)-226 as a surrogate for background uranium when the contaminant is processed uranium. Section 4.3.2 of MARSSIM contains more information on the use of surrogate radionuclides.

The licensee may use multiple reference areas if the reference areas have significantly different background levels because of the variability in background between areas (see Section A.4.4 below and Section 13.2 of NUREG-1505, “A Proposed Nonparametric Statistical Methodology for the Design and Analysis of Final Status Decommissioning Surveys–Interim Draft Report for Comment and Use,” issued June 1998). The licensee may use a derived reference area to extract background information from the survey unit, because a suitable reference area is not readily available. For example, it may be possible to derive a background distribution based on areas of the survey unit where residual radioactivity is not present.

### A.4.2 Characteristics of Soil Reference Areas

The objective is to select nonimpacted background reference areas where the distribution of measurements should be the same as would be expected in the survey unit if that survey unit had never been contaminated. Section 4.5 of MARSSIM contains an acceptable method for selecting background areas, as briefly described below.

Reference areas should have a soil type as similar to the soil type in the survey unit as possible. If there is a choice of possible reference areas with similar soil types, consideration should be given to selecting reference areas that are most similar in terms of other physical, chemical, geological, and biological characteristics. Each reference area should have an area at least as large as the survey unit, if practical, to include the full potential spatial variability in background concentrations. Reference areas may be off site or on site, as long as they are nonimpacted. NUREG-1506, "Measurement Methods for Radiological Surveys in Support of New Decommissioning Criteria, Draft Report for Comment," issued August 1995, provides additional information on reference area selection. Licensees should contact the NRC staff when they are unable to find a reference area that satisfies the above criteria.

### A.4.3 Different Materials in a Survey Unit

Survey units may contain a variety of materials with markedly different backgrounds. An example might be a room with drywall walls, concrete floor, glass windows, metal doors, wood trim, and plastic fixtures. It is not appropriate to make each material a separate survey unit, because the dose modeling is based on the dose from the room as a whole and because a large number of survey units in a room would require an inappropriate number of samples.

When there are different materials with substantially different backgrounds in a survey unit, the licensee may use a reference area that is a nonimpacted room with roughly the same mix of materials as the survey unit.

If a survey unit contains several different materials, but one material is predominant, or if there is not too great a variation in background among materials, a background from a reference area containing only a single material may still be appropriate. For example, a room may be mostly concrete but with some metal beams, and the residual radioactivity may be mostly on the concrete. In this situation, where the concrete predominates, it would be acceptable to use a reference area that contained only concrete. However, the licensee should demonstrate that the selected reference area will not result in underestimating the residual radioactivity on other materials.

The licensee may also use measured backgrounds for different materials or for groups of similar materials. When the licensee decides to use different measured backgrounds for different materials or for a group of materials with similar backgrounds, it is acceptable to perform a one-sample test on the difference between the paired measurements from the survey unit and from the appropriate reference material. Chapter 12 of NUREG-1505 describes, in detail, an acceptable method to do this.

For onsite materials, present either in buildings or as nonsoil materials present in outdoor survey units (e.g., concrete, brick, drywall, fly ash, petroleum product wastes), the licensee should attempt to find nonimpacted materials that are as similar as possible to the materials on the site. Sometimes such materials will not be available. In those situations, the licensee

should make a good faith effort to find the most similar materials readily available or use appropriate published estimates.

### A.4.4 Differences in Backgrounds between Areas

When using a single reference area, any difference in the mean radionuclide concentration between the survey unit and the reference area would be interpreted as caused by residual radioactivity from site operations. This interpretation may not be appropriate when the variability in mean background concentrations among different reference areas is a substantial fraction of the $DCGL_W$. When there may be a significant difference in backgrounds between different areas, the licensee can conduct a Kruskal-Wallis test, as described in Chapter 13 of NUREG-1505, to determine whether there are, in fact, significant differences in mean background concentrations among potential reference areas.

While NUREG-1505 does not recommend specific values for the Kruskal-Wallis test, information on the power of the F-test (parametric complement to the nonparametric Kruskal-Wallis test) is provided in Table 13.5 of NUREG-1505 to assist with the selection of the number of reference areas and measurements per reference area, as part of the DQO process. For example, Table 13.5 of NUREG-1505 indicates that 4 reference areas and 10 to 20 measurements per reference area should generally be adequate. Also, NUREG-1505 states that if four reference areas are selected, an alpha error of 0.1 would be a reasonable default, and in some circumstances larger alpha error values could be considered. NUREG-1505 notes that different alpha and beta error values may be justified based on risk considerations. For example, a lower beta error value than alpha error value would reduce the risk of not detecting background variations that are actually present.

If there are significant differences in backgrounds among reference areas, a value of three times the standard deviation of the mean of the reference area background values could be added to the mean of the reference area background to define a concentration that is distinguishable from background. A value of three times the standard deviation of the mean is chosen to minimize the likelihood that a survey unit that contains only background would fail the statistical test for release. This value can be used as the LBGR in a two-sample test (WRS test) (see Appendix A and G, Sections A.5 and Section G.6) to test whether the survey unit meets the radiological criteria for license termination. Chapters 6 and 13 of NUREG-1505 describes this method in detail.

### A.4.5 Background Survey Design

This survey measures nonimpacted areas on and surrounding the site to establish the baseline; that is, the normal background levels of radiation and radioactivity. In some situations, historical measurements may be available from surveys performed before the construction and operation of a facility. The survey should avoid areas such as roads, parking lots, and other large paved surfaces that may have been impacted or disturbed by site-related activities. The background survey takes on added importance, since the licensee may decide to use a statistical test that compares impacted areas to offsite or onsite reference areas to demonstrate compliance with the release criteria in 10 CFR Part 20, Subpart E, “Radiological Criteria for License Termination.” To minimize systematic biases in the comparison, the same sampling procedure, measurement techniques, and type of instrumentation (e.g., detection sensitivity and accuracy) should be used at both the survey unit and the reference area.

NUREG-1505 provides additional guidance on survey design, the methods of accounting for background radiation, and the nonparametric statistical methods for testing compliance with the decommissioning criteria in 10 CFR Part 20, Subpart E. Formulas contained in NUREG-1505 can be used to compute the required number of samples (measurement points) that will be needed in both the background reference and survey areas.

## A.5 Methods to Evaluate Survey Results

All survey units should be evaluated to determine whether the average concentration in the survey unit as a whole is below the $DCGL_W$. If the radionuclide is not present in background, and the measurement technique is radionuclide-specific so that a comparison with a reference area is not necessary, the licensee should use a one-sample test, the Sign test. Section 8.3 of MARSSIM describes this test.

When the residual radioactivity contains a radionuclide present in the environment or when the measurements are not radionuclide specific, the survey unit should be compared to a reference area. When comparing the survey unit to a reference area, the licensee should use a two-sample test, the WRS test. Section 8.4 of MARSSIM describes this test.

### A.5.1 A Case for not Subtracting Background

An exception to using a two-sample test when a radionuclide is present in background is when the licensee plans to assume that all the radionuclide activity in the survey unit is caused by licensed operations and none is from background. This could be the case for cesium (Cs)-137, for example, because the levels in the environment are often so much less than the $DCGL_W$ that background concentrations may be ignored.

### A.5.2 Elevated Measurements Comparison

Class 1 survey units that pass the Sign test or WRS test but have small areas with concentrations exceeding the $DCGL_W$ should also be tested to demonstrate that those small areas meet the dose criteria for license termination. This test is called the elevated measurement comparison (EMC). It is described in Section 8.5.1 of MARSSIM and summarized here.

To perform the EMC, the licensee first determines the size of the area in the survey unit with a concentration greater than the $DCGL_W$, then determines the $DCGL_{EMC}$ for an area of that size. (The $DCGL_{EMC}$ is the concentration permitted in a limited area of a survey unit; see Section A.8.6.) The average concentration in the area is calculated for comparison against the $DCGL_{EMC}$. The EMC is acceptable if the following condition is met, as shown in Equation A-1 (modified from MARSSIM Equation 8-2):

$$\frac{\delta}{DCGL_W} + \frac{average\ concentration\ in\ the\ elevated\ area - \delta}{DCGL_{EMC}} < 1 \qquad \text{(A-1)}$$

Where δ = the average residual radioactivity concentration for all sample points in the survey unit only.

If there is more than one elevated area, a separate term should be included for each one. As an alternative to the unity rule expressed in Equation A-1, the licensee can calculate the dose from the actual distribution of residual radioactivity, if an appropriate exposure pathway model is available.

## A.6 Instrument Selection and Calibration

To demonstrate that the radiological criteria for license termination have been met, the measurement instruments should have an adequate sensitivity, be calibrated properly, and be checked periodically for proper response.

### A.6.1 Calculation of Minimum Detectable Concentrations

The licensee should determine the MDC for the instruments and techniques to be used. The MDC is the concentration that a specific instrument and technique can be expected to detect 95 percent of the time under actual conditions of use.

For scanning building surfaces for beta and gamma emitters, the $MDC_{scan}$ should be determined from the following equation (obtained by combining MARSSIM Equations 6-8, 6-9, and 6-10 and using a value recommended in this appendix for the index of sensitivity $d'$ of 1.38, which is for 95 percent detection of a concentration equal to $MDC_{scan}$ with a 60 percent false-positive rate).

$$MDC_{scan}\left(building\ surfaces\right) = \frac{270{,}000 \times 1.38\sqrt{B}}{\sqrt{p}\ \varepsilon_i\ \varepsilon_s\ A\,t} \qquad \text{(A-2)}$$

where $MDC_{scan}$ = minimum detectable concentration for scanning building surfaces in picocuries per square meter (pCi/m$^2$)
270,000 = conversion factor to convert to pCi/m$^2$
1.38 = index of sensitivity $d'$
$B$ = number of background counts in time interval $t$
$p$ = surveyor efficiency
$\varepsilon_i$ = instrument efficiency for the emitted radiation
$\varepsilon_s$ = source efficiency in emissions/disintegrations
$A$ = probe's sensitive area in cm$^2$
$t$ = time interval of the observation while the probe passes over the source, in seconds

Based on the measurements described in NUREG/CR–6364, "Human Performance in Radiological Survey Scanning," issued August 2000, a surveyor efficiency $p$ of 0.5 represents a mean value for normal field conditions and its use is generally acceptable. If the licensee wants to determine a value appropriate for particular measurement techniques, the information in NUREG/CR–6364 describes how to determine the value. NUREG-1507, "Minimum Detectable Concentrations with Typical Radiation Survey Instruments for Various Contaminants and Field Conditions," issued June 1998, contains additional information on the interpretation of results reported in NUREG/CR–6364.

For scanning soil with a sodium iodide gamma detector, the $MDC_{scan}$ values given in Table 6.7 of MARSSIM provide an acceptable estimate of $MDC_{scan}$.

For static measurements of surface concentrations, the $MDC_{static}$ may be calculated using the following equation (from Equation 3-10 in NUREG-1507).

$$MDC_{static} = \frac{3+4.65\sqrt{B}}{K\,t} \quad \text{(A-3)}$$

where $MDC_{static}$ = minimum detectable concentration in pCi/m$^2$ or pCi/gram (g)
$B$ = background counts during measurement time interval $t$
$t$ = counting time in seconds
$K$ = a calibration constant (best estimate) to convert counts/second to pCi/m$^2$ or pCi/g, discussed further in NUREG-1507

Section 6.7.1 of MARSSIM shows an example using this equation.

The instruments used for sample measurements at the specific sample locations should have an $MDC_{static}$ less than 50 percent of the $DCGL_W$, as recommended in Section 4.7.1 of MARSSIM. There is no specific recommendation for the $MDC_{scan}$, but the $MDC_{scan}$ will determine the number of samples needed, as discussed in Section A.8.6 of this appendix.

The licensee should record all numerical values measured, even values below the "MDC" or "critical level," including values that are negative (when the measured value is below the average background). Entries for measurement results should not be "nondetect," "below MDC," or similar entries, because the statistical tests can only tolerate a maximum of 40 percent nondetects.

### A.6.2 Instrument Calibration and Response Checks

NRC regulations at 10 CFR 20.1501(c) require that the licensee periodically calibrate radiation measurement instruments used in surveys such as the FSS.

For *in situ* gamma measurements, the detector efficiency (count rate per unit fluence rate) should be determined for the gamma energies of interest and the assumed representative depth distribution. The surface and volumetric distributions should be explicitly considered to evaluate potential elevated areas. To calibrate for the representative depth distribution, acceptable methods are to (1) use a test bed with radioactive sources distributed appropriately or (2) use primarily theoretical calculations that are normalized or verified experimentally using a source approximating a point source. The calibration of the source used for the verification source should be traceable to a recognized standards or calibration organization, for example, the National Institute of Standards and Technology.

Some modern instruments are very stable in their response. Thus, as long as the licensee periodically performs instrument response checks to verify that the detector is operating properly, it may be acceptable to calibrate only initially without periodic recalibrations. The initial calibration may be performed by either the instrument supplier or the licensee, but in either case, 10 CFR 20.2103(a) requires that a record describing the calibration be available for NRC inspection.

### A.6.3 Instrument Response Checks

Licensees should check the response of survey instruments with a check source each day before use to confirm constancy in instrument response and establish criteria for acceptable

response. If the response is not acceptable, the licensee should consider the instrument as not responding properly and should not use it until the problem has been resolved. Measurements made after the last acceptable response check should be evaluated and discarded, if appropriate.

The check source should emit the same type of radiation (i.e., alpha, beta, gamma) as the radiation being measured and should give a similar instrument response, but the check source does not have to use the same radionuclide as the radionuclide being measured.

## A.7 Scanning Coverage Fractions and Investigation Levels

Scanning is performed to locate small areas of elevated concentrations of residual radioactivity to determine whether they meet the radiological criteria for license termination. The licensee should perform scanning in each survey unit to detect areas of elevated concentrations. The licensee should establish ILs for investigating significantly elevated concentrations of residual radioactivity. Table A.2 shows acceptable scanning coverage fractions and scanning ILs for buildings and land areas. This table is based on MARSSIM Roadmap Tables 2 and 5.8.

Systematic scans are those conducted according to a preset pattern. Judgmental scans are those conducted to include areas with a greater potential for residual radioactivity. In Class 2 areas, a 10 percent scanning coverage would be appropriate when there is high confidence that all locations would be below the $DCGL_W$. A coverage of 25 percent to 50 percent would be appropriate when there may be locations with concentrations near the $DCGL_W$. A coverage of 100 percent would be appropriate, if there is any concern that the area should have had a Class 1 classification rather than a Class 2 classification. In Class 3 areas, scanning coverage is usually less than 10 percent. If any location exceeds the scanning IL, scanning coverage in the vicinity of that location should be increased to delineate the elevated area.

**Table A.2 Scanning Coverage Fractions and Scanning Investigation Levels**

| Class | Scanning Coverage Fraction | Scanning Investigation Levels |
|---|---|---|
| 1 | 100 % | >$DCGL_{EMC}$ |
| 2 | 10 to 100 % for soil and for floors and lower walls of buildings, 10 to 50 % for upper walls and ceilings of buildings, systematic and judgmental | >$DCGL_W$ or >$MDC_{scan}$ if $MDC_{scan}$ is greater than $DCGL_W$ |
| 3 | Judgmental | >$DCGL_W$ or >$MDC_{scan}$ if $MDC_{scan}$ is greater than $DCGL_W$ |

Sometimes the sensitivity of static measurements at designated sample points is high enough to detect significantly elevated areas between sample points. If the sensitivity is high enough, only this single set of measurements is necessary. For example, both scanning and sampling for cobalt-60, which emits an easily detectable gamma, can be done with a single set of *in situ* measurements in some cases.

## A.8 Determining the Number of Samples Needed

A minimum number of samples are needed to obtain sufficient statistical confidence that the conclusions drawn from the samples are correct. The method described below from Chapter 5 of MARSSIM is acceptable for determining the number of samples needed.

### A.8.1 Determination of the Relative Shift

The required number of samples will depend on a ratio involving the concentration to be measured relative to the variability in the concentration. The ratio to be used is called the relative shift, $\Delta/\sigma_s$. The relative shift, $\Delta/\sigma_s$, is defined in Section 5.5.2.2 of MARSSIM as:

$$\Delta/\sigma_s = \frac{DCGL_W - LBGR}{\sigma_s} \qquad \text{(A-4)}$$

where $DCGL_W$ = derived concentration guideline
$LBGR$ = concentration at the lower bound of the gray region. The $LBGR$ is the concentration to which the survey unit must be cleaned to have an acceptable probability of passing the test (i.e., 1−β).
$\sigma_s$ = an estimate of the standard deviation of the concentration of residual radioactivity in the survey unit (which includes real spatial variability in the concentration as well as the precision of the measurement system)

The value of $\sigma_s$ is determined either from existing measurements or by taking limited preliminary measurements of the concentration of the residual radioactivity in the survey unit at about 5 to 20 locations, as recommended in Section 5.5.2.2 of MARSSIM. If a reference area will be used and the estimate of the standard deviation in the reference area, $\sigma_r$, is larger than the estimate

of the standard deviation in the survey unit, $\sigma_s$, then the larger value should be used in the equation.

The *LBGR* should be set at the mean concentration of residual radioactivity that is estimated to be present in the survey unit. However, if no other information is available on the survey unit, the *LBGR* may be initially set equal to 0.5 $DCGL_W$, as recommended by MARSSIM. If the relative shift, $\Delta/\sigma_s$, exceeds 3, the *LBGR* should be increased until $\Delta/\sigma_s$ is equal to 3. The licensee may refer to Section 5.5.2.2 of MARSSIM for additional details and information.

### A.8.2 Determination of Acceptable Decision Errors

A decision error is the probability of making an error in the decision on a survey unit by failing a survey unit that should pass or by passing a survey unit that should fail. When using the statistical tests, larger decision errors may be unavoidable when encountering difficult or adverse measuring conditions. This is particularly true when trying to measure residual radioactivity concentrations close to the variability in the concentration of those materials in natural background.

The α decision error is the probability of passing a survey unit where the actual concentration exceeds the release criterion. A decision error α of 0.05 is acceptable under the more favorable conditions when the relative shift, $\Delta/\sigma_s$, is large (about 3 or greater). Larger values of α may be considered when the relative shift is small, to avoid an unreasonable number of samples. The β decision error is the probability of failing a survey unit where the actual concentration is equal to *LBGR*. Any value of β is acceptable to the NRC.

### A.8.3 Number of Samples Needed for the Wilcoxon Rank Sum Test

The minimum number of samples, *N,* needed in each survey unit for the WRS test may be determined from the following equation (adapted from MARSSIM Equation 5-1 with *N* redefined as the number of samples in the survey unit):

$$N = \frac{1}{2} \times \frac{\left(Z_{1-\alpha} + Z_{1-\beta}\right)^2}{3\left(P_r - 0.5\right)^2} \qquad \text{(A-5)}$$

where $N$ = the number of samples in the survey unit
$Z_{1-\alpha}$ = the percentile represented by the decision error α
$Z_{1-\beta}$ = the percentile represented by the decision error β
$P_r$ = the probability that a random measurement from the survey unit exceeds a random measurement from the background reference area by less than the $DCGL_W$ when the survey unit median is equal to the *LBGR* concentration above background
½ = a factor added to MARSSIM Equation 5-1 because *N* is always defined in this guide as the number of samples in the survey unit

Tables 5.1 and 5.2 of MARSSIM contain values of $P_r$, $Z_{1-\alpha}$, and $Z_{1-\beta}$. *N* is the minimum number of samples necessary in each survey unit. An additional *N* samples will also be needed in the reference area. If *N* is not an integer, the number of samples is determined by rounding up. In addition, the licensee should consider taking some additional samples (MARSSIM recommends 20 percent) to protect against the possibility of lost or unusable data. Fewer samples increase

the probability of an acceptable survey unit failing to demonstrate compliance with the radiological criteria for release.

### A.8.4 Number of Samples Needed for Sign Test

The number of samples *N* needed in a survey unit for the Sign test may be determined from the following equation (adapted from MARSSIM Equation 5-2):

$$N = \frac{\left(Z_{1-\alpha} + Z_{1-\beta}\right)^2}{4\left(Sign\,p - 0.5\right)^2} \qquad \text{(A-6)}$$

where $N$ = number of samples needed in a survey unit
$Z_{1-\alpha}$ = percentile represented by the decision error α
$Z_{1-\beta}$ = percentile represented by the decision error β
*Sign p* = estimated probability that a random measurement for the survey unit will be less than the $DCGL_W$ when the survey unit median concentration is actually at the LBGR.

Tables 5.2 and 5.4 of MARSSIM contain the values of $Z_{1-\alpha}$, $Z_{1-\beta}$, and *Sign p*. In addition, the licensee should consider taking some additional samples (MARSSIM recommends 20 percent) to protect against the possibility of lost or unusable data. Fewer samples increase the probability of an acceptable survey unit failing to demonstrate compliance with the radiological criteria for release. If a survey unit fails to demonstrate compliance because there were not enough samples taken, a totally new sampling effort may be needed unless resampling was anticipated.

### A.8.5 Use of Two-Stage or Double Sampling

It may be desirable for a licensee to sample a survey unit a second time to determine compliance. "Two-stage sampling" and "double sampling" are two methods by which additional survey unit data can be acquired. Two-stage sampling refers to survey designs specifically intended to be conducted in two stages. Double sampling refers to the case when the survey unit design is a one-stage design, but allowance is made for a second set of samples to be taken if the retrospective power of the test using the first set of samples does not meet the design objectives. Use of either method should be considered as part of the DQO process when developing the design of the FSS. Appendix C of this volume contains information on the use of two-stage or double sampling.

### A.8.6 Additional Samples for Elevated Measurement Comparison in Class 1 Areas

Additional samples may be needed when the concentration that can be detected by scanning, $MDC_{scan}$, is larger than the $DCGL_W$. The licensee should determine whether additional samples are needed in Class 1 survey units for the EMC when the concentration that can be detected by scanning, $MDC_{scan}$, is larger than the $DCGL_W$. The method in Section 5.5.2.4 of MARSSIM to determine whether additional samples are needed is acceptable to the NRC staff and is described here.

The area factor is the multiple of the $DCGL_W$ that is permitted in a limited portion of the survey unit. In Equation A-7, the ratio of the $MDC_{scan}$ to the $DCGL_W$ establishes the area factor (the

multiple of the $DCGL_W$) that can be detected by scanning (adapted from MARSSIM Equation 5-4):

$$area\ factor = \frac{MDC_{scan}}{DCGL_W} \quad \text{(A-7)}$$

Using the methods in NUREG-1549, "Decision Methods for Dose Assessment To Comply with Radiological Criteria for License Termination, Draft Report for Comment," issued July 1998, the licensee can determine the size of the area corresponding to the area factor, $A_{EC}$. The number of sample points that may be needed to detect this area of elevated measurement concentration, $N_{EMC}$, in a survey unit is:

$$N_{EMC} = \frac{A}{A_{EC}} \quad \text{(A-8)}$$

where $A$ = the area of the survey unit
$A_{EC}$ = the area of concentration greater than $DCGL_W$

If $N_{EMC}$ is larger than $N$, additional samples may be needed to demonstrate that areas of elevated concentrations meet the radiological criteria for license termination. However, the number of samples needed is not necessarily $N_{EMC}$. The licensee can use the HSA and site characterization to determine how many additional samples may be needed. Based on what is known about the site, it may be possible to estimate a concentration that is unlikely to be exceeded. If there is a maximum concentration, the size of the area corresponding to this area factor for this concentration may be used for $A_{EC}$ in Equation A-8. Similarly, based on knowledge of how the radioactive material was handled or dispersed on the site, it may be possible to estimate the smallest area likely to have elevated concentrations. If this is so, that area can be used in Equation A-8. Likewise, the licensee could use the knowledge of how the residual radioactivity would be likely to spread or diffuse after deposition to determine an area $A_{EC}$ for Equation A-8.

Figure D-7 of Appendix D to MARSSIM and Section 3.7.2 of NUREG-1505 show that a triangular grid is slightly more effective in locating areas of elevated concentrations. Therefore, a triangular grid generally should be used if $N_{EMC}$ is significantly larger than $N$ and if areas similar in size or smaller than the grid spacing are expected to have concentrations at or above the area factor.

## A.9 Determining Sample Locations

For the impacted areas, the licensee should establish a reference coordinate system, which is a set of intersecting lines referenced to a fixed site location or benchmark. Reference coordinate systems are established so that the locations of any point in the survey unit can be identified by coordinate numbers. A reference coordinate system does not establish the number of sample points or determine where samples are taken. A single reference coordinate system may be used for a site, or different coordinate systems may be used for each survey unit or for a group of survey units. Section 4.8.5 of MARSSIM describes an acceptable method to establish a reference coordinate system.

In Class 1 and Class 2 areas, the sampling locations are established in a regular pattern, either square or triangular. The method described below is from Section 5.5.2.5 of MARSSIM.

After the number of samples needed in the survey unit has been determined, and the licensee has decided whether to use a square or triangular grid, sample spacings, *L*, are determined from Equations A-9 and A-10 (adapted from MARSSIM Equations 5-5, 5-6, 5-7, and 5-8).

$$L = \sqrt{\frac{A}{0.866\,N}} \quad \textit{for a triangle grid} \tag{A-9}$$

$$L = \sqrt{\frac{A}{N}} \quad \textit{for a square grid} \tag{A-10}$$

where $A$ = the survey unit area
$N$ = the number of samples needed (in Class 1 areas, the larger of the number for the statistical test or the EMC).

The calculated value of *L* is then often rounded downward to a shorter distance that is easily measured in the field.

A random starting point should be identified for the survey pattern. The coordinate location of the random starting point should be determined by a set of two random numbers, with one representing the *x* axis and the other, the *y* axis. The random numbers can be generated by calculator or computer or can be obtained from a table of random numbers. Each random number should be multiplied by the appropriate survey unit dimension to provide a coordinate relative to the origin of the survey unit reference coordinate system.

Beginning at the random starting point, a row of points should be identified parallel to the *x* axis at intervals of *L*. For a square grid, the additional rows should be parallel to the first row at a distance of *L* from the first row. For a triangular grid, the distance between rows should be 0.866 *L*, and the sample locations in the adjacent rows should be midway on the *x* axis between the sample locations in the first row. Sample locations selected in this manner that either do not fall within the survey unit area or cannot be surveyed because of site conditions should be replaced with other sample locations determined using the same random selection process that was used to select the starting point. MARSSIM Figure 5.5 contains an example illustrating the triangular grid pattern.

In Class 3 survey units and in reference areas, all samples should be taken at random locations. Each sample location should be determined by a set of two random numbers, one representing the *x* axis and the other the *y* axis. Each set of random numbers should be multiplied by the appropriate survey unit dimension to provide coordinates relative to the origin of the survey unit reference coordinate system. Coordinates identified in this manner that do not fall within the survey unit area or that cannot be surveyed because of site conditions should be replaced with other sample locations determined in the same manner. MARSSIM Figure 5.4 illustrates a random sample location pattern.

## A.10 Determining Compliance

The licensee should first review the measurement data to confirm that the survey units were properly classified. MARSSIM Section 8.2.2 contains methods for this review that are acceptable to the NRC staff. If the FSS shows that an area was misclassified with a less restrictive classification, the area should receive the correct classification and the FSS for the area should be repeated. A pattern of misclassifications that are not restrictive enough indicates that the characterization was not adequate. In this case, the site or portions of the site in question should be characterized again, reclassified, and resurveyed for the new classification. The licensee should then determine whether the measurement results demonstrate that the survey unit meets the radiological criteria for license termination. Tables A.3 and A.4, below, summarize an acceptable way to interpret the sample measurements. MARSSIM Section 8.4 describes the WRS test, while MARSSIM Section 8.3 describes the Sign test and MARSSIM Section 8.5 describes the EMC. The elevated measurement is applied to all sample measurements and all scanning results that exceed the $DCGL_W$.

In some cases, licensees may choose to use scanning or fixed measurement techniques that assess 100 percent of the population of potential direct measurements or samples within the survey unit. For these cases, it may be reasonable to demonstrate compliance by directly comparing the average radionuclide concentrations determined from the survey with the appropriate $DCGL_W$, without the need to perform statistical tests. Guidance has not yet been developed for using such techniques without performing statistical tests; therefore, licensees should discuss such techniques with the NRC staff on a case-by-case basis.

**Table A.3 Interpretation of Sample Measurements when a Reference Area is Used**

| Measurement Results | Conclusion |
|---|---|
| Difference between maximum survey unit concentration and minimum reference area concentration is less than $DCGL_W$. | Survey unit meets release criterion. |
| Difference between survey unit average concentration and reference area average concentration is greater than $DCGL_W$. | Survey unit fails. |
| Difference between any survey unit concentration and any reference area concentration is greater than $DCGL_W$ and the difference of survey unit average concentration and reference area average concentration is less than $DCGL_W$. | Conduct WRS test and EMC. |

**Table A.4 Interpretation of Sample Measurements when no Reference Area is Used**

| Measurement Results | Conclusion |
|---|---|
| All concentrations are less than $DCGL_W$. | Survey unit meets release criterion. |
| Average concentration is greater than $DCGL_W$. | Survey unit fails. |
| Any concentration is greater than $DCGL_W$ and average concentration less than $DCGL_W$. | Conduct Sign test and EMC. |

Some facilities may have residual radioactivity composed of more than one radionuclide. When there are multiple radionuclides rather than a single radionuclide, the licensee should consider the dose contribution from each radionuclide. Section 2.7 of this volume contains information about using the sum of fractions approach for compliance when multiple radionuclides are present.

When there is a fixed ratio among the concentrations of the nuclides, a $DCGL_W$ for each nuclide can be calculated. Compliance with the radiological criteria for license termination may be demonstrated by comparing the concentration of the single surrogate radionuclide that is easiest to measure with its $DCGL_W$ (which has been modified to account for the other radionuclides present). For example, if Cs-137 and Sr-90 are present, using measured concentrations of Cs-137 as a surrogate for the mix of Cs-137 and Sr-90 may be simpler than separately measuring Cs-137 and Sr-90, and may thus save labor and analytical expenses. When using a surrogate radionuclide to represent the presence of other radionuclides, a sufficient number of measurements, spatially distributed throughout the survey unit, should be used to establish a consistent ratio between the surrogate and the other radionuclides. Section 4.3.2 of MARSSIM provides additional information on the use of surrogate radionuclides for surveys.

When there is no fixed ratio among the concentrations of the nuclides, the licensee must evaluate the concentration of each nuclide via sampling. Compliance with the radiological criteria for license termination is then demonstrated by considering the sum of the concentration of each nuclide relative to its $DCGL_W$, followed by an evaluation of all radionuclides of concern via the unity rule. Chapter 11 of NUREG-1505 describes an acceptable method for performing the evaluation. When there is no fixed ratio among the concentrations of the nuclides, and a large number of discrete samples are required, it may be possible to utilize a composite sampling strategy to increase the probability of elevated area detection and as a means to reduce analytical cost. However, this approach requires an evaluation of the exposure scenario related to the hard-to-detect radionuclide and should be performed on a case-by-case basis along with discussions with the regulator. Additional information on composite sampling can be found in Appendix O of this volume.

In some cases in which multiple nuclides are present with no fixed ratio in their concentrations, the dose contribution from one or more of the nuclides in the mixture will dominate the total dose, and the dose from other radionuclides will be insignificant. For example, at a nuclear power plant, many different radionuclides could be present with no fixed ratio in their

concentrations, but almost all of the dose would come from just one or two of the nuclides. Section 3.3 of this volume contains guidance on elimination of radionuclides or pathways from consideration.

If a survey unit fails, the licensee should evaluate the measurement results and determine why it failed. MARSSIM, in Sections 8.2.2 and 8.5.3, and in Appendix D, provides acceptable methods for reviewing measurement results. If it appears that the failure was caused by the presence of residual radioactivity in excess of that permitted by the radiological release criteria, the survey unit should be re-remediated and resurveyed. However, some failures may not be caused by the presence of residual radioactivity. If it can be determined that this is the case, the survey unit may be released.

## A.11 References


Nuclear Regulatory Commission (U.S.) (NRC). "Measurement Methods for Radiological Surveys in Support of New Decommissioning Criteria, Draft Report for Comment," NUREG-1506. NRC: Washington, DC. August 1995.

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— — — — —. "Human Performance in Radiological Survey Scanning," NUREG/CR-6364. NRC: Washington, DC. 1998b.

— — — — —. "Minimum Detectable Concentrations with Typical Radiation Survey Instruments for Various Contaminants and Field Conditions," NUREG-1507. NRC: Washington, DC. 1998c.

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— — — —. "Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)," NUREG-1575, Rev. 1. EPA 402-R-97-016, Rev. 1; DOE/EH-0624, Rev. 1; U.S. Department of Defense; U.S. Department of Energy; U.S. Environmental Protection Agency; and NRC: Washington, DC. August 2000.

— — — — —. Regulatory Issue Summary 2002–02, "Lessons Learned Related to Recently Submitted Decommissioning Plans and License Termination Plans." NRC: Washington, DC. January 2002.

# APPENDIX B

# SIMPLE APPROACHES FOR CONDUCTING FINAL RADIOLOGICAL SURVEYS

## B.1 Introduction

A large number of licensees may use a simplified method to demonstrate regulatory compliance for decommissioning, and thereby avoid complex FSSs. For Decommissioning Groups 1–3, licensees may use the simplified FSS method described in Appendix B to NUREG-1575, "Multi-Agency Radiological Survey and Site Investigation Manual" (MARSSIM), Revision 1, issued August 2000, or the alternative protocol described in this appendix (Section B.3 below).

## B.2 MARSSIM Simplified Method

The simplified method in Appendix B of MARSSIM may be used by Decommissioning Group 1 and some Decommissioning Group 2 licensees. These are sites where radioactive materials have been used or stored only in the form of (1) nonleaking, sealed sources, (2) short half-life radioactive materials (e.g., $T_{1/2} \leq 120$ days) that have since decayed to insignificant quantities, (3) small quantities exempted or not requiring a specific license, or (4) a combination of the above. MARSSIM, Revision 1, Appendix B gives the details of this simplified method.

## B.3 Alternative Simplified Method

This alternative method may be used by Decommissioning Groups 1–3 and is applicable only for surfaces of building structures and for surface soils. The following conditions are prerequisite to the use of this method:

- screening values are applicable and being used to demonstrate compliance with release criteria (i.e., site meets underlying assumptions in screening analyses discussed in more detail in Appendix H)
- removable residual radioactivity on building surfaces 10 percent or less; or adjusted to account for higher removable fractions as discussed in Appendix H of this volume
- no sources requiring complex or special surveys are present (e.g., no (i) volumetric building structure residual radioactivity, (ii) duct work, (iii) embedded piping, (iv) groundwater residual radioactivity, (v) subsurface soil residual radioactivity, (vi) buried conduit, (vii) sewer pipes, or (viii) prior onsite disposals)
- not to be applied to land areas where soil has been previously remediated
- minimum detectable concentrations between 10 and 50 percent of the $DCGL_W$ for scans, static or direct measurements, and sampling and analysis (using guidance in NUREG-1507, "Minimum Detectable Concentrations with Typical Radiation Survey Instruments for Various Contaminants and Field Conditions")[1]

If the above conditions are met, then the following simplified method may be used to design and conduct the FSS for each survey unit.

- Size is limited to 2000 $m^2$ for land areas and 100 $m^2$ for structures.

[1] Revision 1 of NUREG-1507 was issued in August 2020.

- Scanning and sampling are to be performed as follows:
  - 100-percent scan
  - 30 samples
- The hot spot criterion is three times the $DCGL_W$, applied to any sampling location.
- A quality control program ensures results are accurate and sources of uncertainty are identified and controlled.
- The average concentration for the survey unit is compared to the $DCGL_W$.
- Statistical tests that may be used include the parametric Student's t-test, or non-parametric WRS test when the radionuclide(s) of concern are in background, or the non-parametric Sign test when the radionuclide(s) of concern are not in background assuming an alpha (α) or false positive error of 5 percent. No statistical tests are needed if all measurements are less than the $DCGL_W$. MARSSIM (NUREG-1575) can be consulted for additional information on the statistical tests.

The FSSR should provide a complete and unambiguous record of the radiological status of the site and should stand on its own with minimal information incorporated by reference (see Appendix D of this volume for additional information on reporting survey results).

## B.4 References

Nuclear Regulatory Commission (U.S.) (NRC). "Minimum Detectable Concentrations with Typical Radiation Survey Instruments for Various Contaminants and Field Conditions," NUREG-1507. NRC: Washington, DC. 1998.

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# APPENDIX C

# USE OF TWO-STAGE OR DOUBLE SAMPLING FOR FINAL STATUS SURVEYS

## C.1 Introduction

This appendix contains information on the assessment of survey data when double or two-stage sampling in a survey unit is used to determine compliance. These sampling strategies utilize the initial sample data and a second round of supplemental sampling. This approach may be desirable when a survey unit fails the hypothesis test (i.e., the decision is made that the survey unit does not meet the release criterion) due to insufficient power as described in more detail below,[1] but the mean of the measured data is below the release criterion. Further information on survey unit failures and possible remedies using the DQO process are discussed in Chapter 8 of NUREG-1575, "Multi-Agency Radiological Survey and Site Investigation Manual" (MARSSIM), Revision 1, issued August 2000.

In this appendix two-stage sampling denotes survey designs specifically intended to be conducted in two stages, whereas double sampling refers to one-stage survey designs that allow for the collection of a second set of samples if the retrospective power of the test using the first set of samples does not meet the design objectives. Double or two -stage sampling may be acceptable if it is incorporated into the final status survey *design*. Thus, before the initial round of sampling occurs, allowances for these sampling approaches should be discussed in the DQOs for the final status survey.

In general, adequate initial sampling to achieve the desired statistical power and error rates, or data collection in two stages, are both preferable to double sampling. As discussed in the next sections of this appendix, there are several factors to consider when additional random samples are collected. For example, double sampling is also generally not appropriate for Class 1 survey units having confirmed areas of elevated activity, or Class 2 or Class 3 survey units because the need for a second set of samples raises the issue of survey unit misclassification.

## C.2 Double Sampling

As noted in the preceding section C.1, situations can occur where a survey unit might have passed the final status statistical test, had the initial sampling design been powerful enough to reject the null hypotheses in Scenario A. That is, a retrospective examination of the power of the statistical tests reveals that the probability of detecting that the survey unit actually meets the release criterion was lower than that planned for during the DQO process. This could occur if the spatial variability in residual radioactivity concentrations was larger than anticipated. The power of the test specified during the DQO process depends on an estimate of the uncertainty. The power of the statistical test will be less than planned if the standard deviation is higher than expected. If samples were lost, did not pass analytical QA/QC, or are otherwise unavailable for inclusion in the analysis, the power will also be lower than was planned. In these situations, it might be desirable to take additional samples in the survey unit to improve the power of the statistical test.

Draft NUREG/CR-5849, "Manual for Conducting Radiological Surveys in Support of License Termination," issued June 1992, allowed the licensee to take additional samples in a survey unit if, after the first sampling, the mean was less than the $DCGL_W$, and the desired upper

[1] Power is the probability of rejecting the null hypothesis when it is false (i.e., in Scenario A, concluding that the site is clean when it is clean). The power is equal to one minus the Type II (false negative) error rate (i.e., (1-β)). The power of the sampling design is important to decision making and can be determined prospectively when planning the survey and retrospectively when interpreting survey results. Additional information on prospective and retrospective power analyses is provided in Chapter 8 and Appendix I of MARSSIM, Revision 1.

confidence level on the mean was greater than the $DCGL_W$. Because a 95-percent confidence interval is constructed using Student's t-statistic rather than using a hypothesis test, the survey design does not consider Type II errors. The second set of samples was taken so that a Student's t-test on the combined set of samples would have 90-percent power at the mean of the first set of samples, given the estimated standard deviation from the first set of samples. Such double sampling was to be performed only once.

Increasing the probability that a clean survey unit passes (power in Scenario A) using double sampling will also tend to increase the probability that a survey unit that is not clean will pass (Type I error). In addition, the two tests are not independent, because the data from the first set of samples is used in both. The increase in the Type I error rate is expected to be less than a factor of two based on the analysis in the following paragraphs of this appendix. The potential increase in the Type I error rate should be considered when designing the survey and determining acceptable error rates as part of the DQO process.

Two-stage or double sampling is not usually expected (nor is it encouraged) when the DQO process is used, as in MARSSIM, Revision 1. This is because the Type II error and the power desired are explicitly considered in the survey design process. If higher power in the test is desired, it should be specified as such. Sufficient samples should be taken to achieve the specified power. The value of this approach lies in the greater objectivity and defensibility of the decision made using the data. Nonetheless, it is recognized that there may be instances when some sort of double sampling is considered desirable. As discussed above, an example is when it is difficult to estimate the standard deviation of the concentrations in a survey unit. A first set of data may be taken with an estimated standard deviation that is too low, and thus, the power specified in the DQO process may not be achieved.

For planning purposes, it is important to understand the circumstances under which it is appropriate to combine this data with additional sample data to be used in the test of the final status survey. Consider the Sign test, as indicated in NUREG-1505, "A Proposed Nonparametric Statistical Methodology for the Design and Analysis of Final Status Decommissioning Surveys–Interim Draft Report for Comment and Use," issued June 1998. Suppose $N_1$ samples are taken. For the Sign test in Scenario A, the test statistic, $S_1$, is equal to the number of survey unit measurements below the $DCGL_W$. If $S_1$ exceeds the critical value $k_1$, then the null hypothesis that the median concentration in the survey unit exceeds the $DCGL_W$ is rejected (i.e., the survey unit passes this test). The probability that any single survey unit measurement falls below the $DCGL_W$ is found from

$$p(C) = \int_{-\infty}^{DCGL_W} f(x)dx = \frac{1}{\sqrt{2\pi}\sigma}\int_{-\infty}^{DCGL_W} e^{\frac{-(x-C)^2}{2\sigma^2}}\,dx = \Phi\left(\frac{DCGL_W - C}{\sigma}\right) \qquad \text{(C-1)}$$

$C$ is the true, but unknown, mean concentration in the survey unit. Assuming the data are normally distributed, when $C = DCGL_W$, then $p = 0.5$. More generally, if C is the true, but unknown, median concentration in the survey unit, p is also equal to 0.5.

The probability that more than $k_1$ of the $N_1$ survey unit measurements fall below the $DCGL_W$ is simply the following binomial probability:

$$\sum_{t=k_1+1}^{N_1}\binom{N_1}{t}p^t\left(1-p\right)^{N_1-t}=1-\sum_{t=0}^{k_1}\binom{N_1}{t}p^t\left(1-p\right)^{N_1-t} \quad \text{(C-2)}$$

This is the probability that the null hypothesis will be rejected, and it will be concluded that the survey unit meets the release criterion. When the median concentration in the survey unit is at the $DCGL_W$, this probability is just the Type I error rate, α. When $C = DCGL_W$, $p = (1- p) = 0.5$, so

$$\alpha = \sum_{t=k_1+1}^{N_1}\binom{N_1}{t}(0.5)^t(0.5)^{N_1-t} = (0.5)^{N_1}\sum_{t=k_1+1}^{N_1}\binom{N_1}{t} \quad \text{(C-3)}$$

Suppose it is decided to allow the licensee to take a second set of samples of size $N_2$. The test statistic, $S$, is equal to the number of the total of $N = N_1 + N_2$ survey unit measurements below the $DCGL_W$. If $S$ exceeds the critical value $k$, then the null hypothesis that the median concentration in the survey unit exceeds the $DCGL_W$ is rejected (i.e., the survey unit passes this test). In this case, the overall probability that the null hypothesis is rejected (i.e., the survey unit passes) is equal to the sum of the probabilities of the two *events*, labeled (Event 1) and (Event 2) below, that are mutually exclusive:

(Event 1) the probability that more than $k_1$ of the $N_1$ survey unit measurements fall below the $DCGL_W$, and

(Event 2) the probability that fewer than $k_1$ of the first $N_1$ survey unit measurements fall below the $DCGL_W$ but that more than $k$ of the $N$ total survey unit measurements fall below the $DCGL_W$.

The test statistic, S, is then equal to the sum of $S_1$ and $S_2$, ($S = S_1 + S_2$), where $S_2$ is the number of the second set of $N_2$ survey unit measurements that fall below the $DCGL_W$. $S_1$ and $S_2$ are independent, but $S_1$ and $S = S_1 + S_2$ are not.

The covariance of $S_1$ and $S$ using $E(\cdot)$ to denote expected values, is

$$\begin{aligned} Cov(S_1,S) &= E(S_1,S) - E(S_1)E(S) \\ &= E(S_1(S_1+S_2)) - E(S_1)E(S) \\ &= E(S_1^2) + E(S_1S_2) - E(S_1)E(S) \\ &= (N_1^2p(1-p) + N_1^2p^2) + N_1N_2p^2 - N_1p(N_1+N_2)p \\ &= N_1p(1-p) \end{aligned} \quad \text{(C-4)}$$

Therefore, the correlation coefficient between $S_1$ and $S$ is

$$\rho(S_1, S) = \frac{\mathrm{N}_1 p(1-p)}{\sqrt{N_1 p(1-p)(N_1+N_2)p(1-p)}}$$

$$= \frac{\mathrm{N}_1}{\sqrt{N_1(N_1+N_2)}}$$

$$= \sqrt{\frac{N_1}{(N_1+N_2}} \qquad \text{(C-5)}$$

$$= \sqrt{\frac{N_1}{N}}$$

To calculate the overall probability that the survey unit passes, one requires the joint probability of $S_1$ and $S$,

$$\begin{aligned}\Pr\left(S_1 = s_1, S = s\right) &= \Pr\left(S_1 = s_1\right)\Pr\left(S_2 = s - s_1\right) \\ &= \binom{N_1}{s_1} p^{s_1}\left(1-p\right)^{N_1 - s_1}\binom{N_2}{s-s_1} p^{s-s_1}\left(1-p\right)^{N_2-(s-s_1)} \\ &= \binom{N_1}{s_1}\binom{N_2}{s-s_1} p^{s}\ \left(1-p\right)^{N-s}\end{aligned} \qquad \text{(C-6)}$$

Therefore, the overall probability that the survey unit passes is

$$\begin{aligned}\Pr\left(S_1 > k_1\ or\ S > k\right) &= \Pr(S_1 > k_1) + \Pr(S_1 \le k_1\ and\ S > k) \\ &= \sum_{s_1 = k_1+1}^{N_1}\binom{N_1}{s_1} p^{s_1}(1-p)^{N_1 - s_1} \\ &+ \sum_{s_1 \le k_1}\ \sum_{s_2 > k - s_1}\binom{N_1}{s_1}\binom{N_1}{s_2} p^{s_1+s_2}(1-p)^{(N_1+N_2)-(s_1+s_2)}\end{aligned} \qquad \text{(C-7)}$$

The first term is equal to (or slightly less than) the Type I error rate α specified during the DQO process. The second term is the additional probability of a Type I error introduced by allowing double sampling.

Note that

$$\Pr\left(S_1 \le k_1 \text{ and } S > k\right) = \sum_{s>k}^{N} p^s (1-p)^{N-s} \sum_{s_1=0}^{k_1} \binom{N_1}{s_1}\binom{N_2}{k-s_1}$$

$$\le \sum_{s>k}^{N} p^s (1-p)^{N-S} \sum_{s_1=0}^{k} \binom{N_1}{s_1}\binom{N_2}{k-s_1} \qquad \text{(C-8)}$$

$$= \sum_{s>k}^{N} p^s (1-p)^{N-s} \binom{N}{s} = \Pr(S > k) \le \alpha$$

Thus, the Type I error rate would be at most doubled when double sampling is allowed.

For example, if a survey is designed so that $N_1$ = 30, and α = 0.05, then the critical value for the Sign test is $k_1$ = 19. Suppose the first survey results in 19 or fewer measurements that are less than the *DCGL*$_W$. In addition, suppose the survey unit is sampled again, taking an additional $N_2$ = 30 samples. Then the total number of samples is $N = N_1 + N_2 = 60$. The critical value for the Sign test with α = 0.05 and $N$ = 60 is $k$ = 36. When the survey unit concentration is equal to the *DCGL*$_W$, $p$ = 0.5, one has

$$\Pr\left(S_1 > 19 \text{ or } S > 36\right) = \Pr\left(S_1 > 19\right) + \Pr\left(S_1 \le 19 \text{ and } S > 36\right)$$

$$= \sum_{s_1=20}^{30} \binom{30}{s_1} (0.5)^{s_1} (1-0.5)^{30-s_1}$$

$$+ \sum_{s_1=0}^{19} \binom{30}{s_1} \sum_{s_2=(37-s_1)}^{30} \binom{30}{s_2} (0.5)^{s_1+s_2} (1-0.5)^{(30+30)-(s_1+s_2)} \qquad \text{(C-9)}$$

$$= 0.049 + 0.027 = 0.076$$

Thus, the total Type I error rate is about 50 percent greater than originally specified.

In conclusion, double sampling should not be used as a substitute for adequate planning. If it is to be used, it should be considered part of the DQO process. The procedure for double sampling (i.e., the size of the second set of samples, $N_2$,) should be specified, recognizing that the Type I error rate could be up to twice that specified for the Sign test when only one set of samples is taken.

Similar considerations apply for the WRS test; however, the calculation of the exact effect on the Type I error rate is considerably more complex and is not discussed in this appendix.

Finally, double sampling should never be necessary for Class 2 or Class 3 surveys, which are not expected to have concentrations above the *DCGL*$_W$. These classes of survey unit should always pass after the first set of samples, because every measurement should be below the *DCGL*$_W$. The need for a second set of samples (i.e., failure to reject the null hypothesis) in Class 2 or Class 3 survey units would raise an issue of survey unit misclassification. In addition, double sampling is generally not appropriate for Class 1 survey units where elevated areas have been found.

In lieu of double sampling, a preferred approach is to plan for data collection in two stages and design the final status survey accordingly, as is discussed in the remainder of this appendix.

## C.3 Two-Stage Sequential Sampling

In some cases, two-stage sampling may be used instead of a single-stage sample design. For example, if there are a large number of survey units of a similar type to be tested, a two-stage sampling procedure may result in substantial savings of time and money by reducing the average number of samples required to achieve a given level of statistical power.

An example of a two-stage sampling design using the Sign test is summarized here. In this example, $N_1$ is the size of the first set of samples taken, and $S_1$ is the number of these samples that are less than the $DCGL_W$. Similarly, $N_2$ is the size of the second set of samples taken, and $S_2$ is the number of these samples that are less than the $DCGL_W$. Let $N = N_1 + N_2$, and let $S = S_1 + S_2$. The procedure is as follows:

- If $S_1 > u_1$ then the survey unit passes (reject $H_0$).
- If $S_1 < l_1$ then the survey unit fails.
- If $l_1 \le S_1 \le u_1$ then the second set of samples is taken.
- If $S = S_1 + S_2 > u_2$ after the second set of samples is analyzed, then the survey unit passes.

An advantage of two-stage sampling is that it can reduce the total number of samples if there are many similar survey units of similar design. For given error rates α and β, the number of samples, $N_1$, taken in the survey unit during the first stage of sampling will be less than the number, $N_0$, required in the MARSSIM, Revision 1, tables. Unless the result is "too close to call," this will be the only sampling needed. When the result is "too close to call," $l_1 \le S_1 \le u_1$, a second sample of size $N_2$ is taken, and the test statistic $S_2$ is computed using the combined data set, $N_1 + N_2$. While the size of the combined set, $N = N_1 + N_2$, will generally be larger than the number, $N_0$, from the MARSSIM, Revision 1, tables, the expected sample size over many survey units will still be lower. Thus, a two-stage sampling scheme will be especially useful when there are many similar survey units for which the FSS design is essentially the same. Two-stage sampling may be used whether or not a reference area is needed (i.e., it may be used with either the Sign or the WRS test).

The remaining major issue is how to choose the critical values $l_1$, $u_1$, and $u_2$. Hewett and Spurrier (1983) suggest three criteria:

(1) Match the power curve of the two-stage test to that of the one-stage test. The curves are matched at three points. The points with power equal to α, 1−β, and 0.5 are generally well enough separated to ensure a good match over the entire range of potential survey unit concentrations.

(2) Maximize the power at the LBGR for given values of α and average sample size.

(3) Minimize the sample size for given values of α, and 1−β.

While any one of these criteria could be used, the first has received more attention in the literature. Thus, it may be more readily applied to the case of final status survey design. The other criteria would require further development.

Spurrier and Hewett (1975) initially developed a two-stage sampling methodology using criteria 1 assuming the data are normally distributed. They matched the two-stage power curve to the one-stage power curve at three points: the first at α values of either 0.05 or 0.01; the second in the gray region where power is equal to 0.5; and the third at the lower bound of the gray region with a β value of 0.1 and power (1-β value) of 0.9. Table C.1 shows the values of $l_1$, $u_1$, and $u_2$ obtained for six different sets of sample sizes, $N_1/N_0$, $N_2/N_0$, expressed as fractions of the sample size, $N_0$, which would be required for the one-stage test with equivalent power. The term $E(N)/N_0$, is the maximum expected combined sample size for the two-stage test relative to the sample size, $N_0$, which would be required for the one-stage test with equivalent power. This number is almost always less than one, but it depends on how close the actual concentration in the survey unit is to the $DCGL_W$. Clearly, if the concentration is over the $DCGL_W$, the survey unit is likely to fail on the first set of samples. If the concentration is much lower than the $DCGL_W$, the survey unit is likely to pass on the first set of samples. It is only when the true concentration in the survey unit falls within the gray region that there will be much need for the second set of samples. The fact that the maximum $E(N)/N_0$ is almost always less than one indicates that the overall number of samples required for a two-stage FSS will almost never exceed the number required for a one-stage test, even if the true concentration of the survey unit falls in the gray region between the *LBGR* and the $DCGL_W$.

The power to distinguish clean from dirty survey units is relatively low when the true concentration is in the gray region because the power falls from 1−β at the *LBGR* to α at the $DCGL_W$. Thus, when the true concentration is in the gray region, a larger number of cases will require a second set of samples. The gray region is exactly where the results are "too close to call." However, if the true concentration of the survey unit is below the *LBGR* or above the $DCGL_W$, the actual average number of samples will be closer to $N_1$, because the second set of samples will seldom be needed.

In 1976, Spurrier and Hewett dropped the assumption of normality and extended their methodology to two-stage Wilcoxon Signed Rank and WRS tests. The procedure depends on an extension of the Central Limit Theorem to the joint distribution of the test statistics $S_1$ and $S = S_1 + S_2$. Spurrier and Hewett suggest that the approximation works reasonably well for sample sizes as small as nine.

In this appendix, their method is also applied to the Sign test.

For the Sign test, one computes

$$S_1 = \frac{S_1^+ - N_1/2}{\sqrt{N_1/4}} \qquad \text{(C-10)}$$

where $S_1^+$ is the usual Sign test statistic (i.e., the number of measurements less than the $DCGL_W$).

Using Table C.1,

- if $S_1 > u_1$ then reject the null hypothesis (the survey unit passes),

- if $S_1 < l_1$ then do not reject the null hypothesis (the survey unit fails),
- if $l_1 \le S_1 \le u_1$ then take the second set of samples.

If a second set of samples is taken, then compute

$$S = \frac{(S_1^+ + S_2^+) - (N_1 + N_2)/2}{\sqrt{(N_1 + N_2)/4}} = \frac{S^+ - N/2}{\sqrt{N/4}} \quad \text{(C-11)}$$

Using Table C.1,

- if $S > u_2$ then reject the null hypothesis (the survey unit passes),
- if $S \le u_2$ then do not reject the null hypothesis (the survey unit fails).

This test relies on "a large sample approximation." That is, one is assuming that the sample size is large enough that the joint distribution of $S_1$ and $S$ is bivariate standard normal with correlation coefficient

$$\rho(S_1, S) = \sqrt{\frac{N_1}{N}} \quad \text{(C-12)}$$

Some simulation studies may be done to determine quantitative bounds on the accuracy of this approximation.

The choice of which set of sample sizes should be used is dependent on how confident one is of passing.

For Class 2 and Class 3 survey units (discussed in Appendix A of this volume), case 3 with $N_1/N_0 = 0.2$ and $N_2/N_0 = 1.0$ might be reasonable. In these classes of survey units, no individual sample concentrations in excess of the $DCGL_W$ are expected. The probability of passing on the first set of samples should be close to one. Therefore, it makes sense to choose a design with the minimum number of samples required in the first set.

For Class 1 survey units (discussed in Appendix A of this volume), case 2 with $N_1/N_0 = 0.4$ and $N_2/N_0 = 0.8$ might be more appropriate. There is some chance that the survey unit will not pass on the first set of samples, so it may be desirable to reduce Max $E(N)/N_0$ from 0.999 to 0.907 by taking more samples in the first set.

If the gray region has been expanded to increase $\Delta/\sigma$, case 1 or 4 would be a more conservative choice. In this situation, statistical power has been compromised somewhat, so it may be important to reduce the risk of having a larger average total number of samples (as indicated by the potential Max $E(N)/N_0$ even further).

Scan sensitivity will also affect the ability to use two-stage designs in Class 1 survey units. It would have to be determined if the $DCGL_{EMC}$ can be detected when only $N_1$ samples are taken. If not, the sample size would have to be increased until the $MDC_{scan}$ is lower than the $DCGL_{EMC}$.

In this situation, the choice of $N_1$, and the average savings possible with two-stage sampling may be severely limited.

**Table C.1 Critical Points for Two-Stage Test of Normal Mean for a One-Sided Alternative**

| | | | α = 0.05 | | | | α = 0.01 | | | |
|---|---|---|---|---|---|---|---|---|---|---|
| | $N_1/N_0$ | $N_2/N_0$ | $u_1$ | $l_1$ | $u_2$ | Max $E(N)/N_0$ | $u_1$ | $l_1$ | $u_2$ | Max $E(N)/N_0$ |
| 1 | 0.6 | 0.6 | 1.886 | 0.71 | 1.783 | 0.866 | 2.499 | 1.259 | 2.493 | 0.879 |
| 2 | 0.4 | 0.8 | 1.984 | 0.179 | 1.782 | 0.907 | 2.558 | 0.635 | 2.496 | 0.931 |
| 3 | 0.2 | 1 | 2.073 | −0.482 | 1.784 | 0.999 | 2.6 | −0.146 | 2.502 | 1.03 |
| 4 | 0.55 | 0.55 | 2.05 | 0.438 | 1.716 | 0.869 | 2.635 | 0.966 | 2.411 | 0.878 |
| 5 | 0.66 | 0.66 | 1.781 | 0.95 | 1.868 | 0.882 | 2.415 | 1.52 | 2.6 | 0.897 |
| 6 | 0.7 | 0.7 | 1.749 | 1.045 | 1.909 | 0.893 | 2.39 | 0.628 | 2.651 | 0.908 |

Source: Spurrier and Hewett (1975).

For the WRS test, at each stage, one sets the number of measurements required in the survey unit, $n_1$ and $n_2$, and in the reference area $m_1$ and $m_2$, relative to the number required for the one-stage test $n_0 = m_0 = N_0/2$ specified in Table 5.3 of MARSSIM, Revision 1. There is an additional requirement that $n_1/n_2 = m_1/m_2$, which should be satisfied with sufficient accuracy for most MARSSIM, Revision 1 designs. Minor departures due to small differences in sample size caused by filling out systematic grids or the loss of a few samples should not severely affect the results.

One now computes

$$S_1 = \frac{W_1^R - m_1\left(n_1 + m_1 + 1\right)/2}{\sqrt{n_1 m_1\left(n_1 + m_1 + 1\right)/12}} \qquad \text{(C-13)}$$

where $W_1^R$ is the usual WRS test statistic (i.e., the sum of the ranks of the adjusted reference area measurements).

Using Table C.1,

- if $S_1 > u_1$ then reject the null hypothesis (the survey unit passes),
- if $S_1 < l_1$ then do not reject the null hypothesis (the survey unit fails),
- if $l_1 \le S_1 \le u_1$ then take the second set of samples.

If a second set of samples is taken, then compute the following.

$$S = \frac{\left(W_1^+ + W_2^+\right) - \left(m_1 + m_2\right)\left(m_1 + m_2 + n_1 + n_2 + 1\right)/2}{\sqrt{\left(m_1 + m_2\right)\left(n_1 + n_2\right)\left(m_1 + m_2 + n_1 + n_2 + 1\right)/12}} = \frac{W^R - m\left(m+n+1\right)/2}{\sqrt{mn\left(m+n+1\right)/12}} \quad \text{(C-14)}$$

Using Table C.1,

- if $S > u_2$ then reject the null hypothesis (the survey unit passes),
- if $S \le u_2$ then do not reject the null hypothesis (the survey unit fails).

This test relies on "a large sample approximation." That is, one is assuming that the sample size is large enough that the joint distribution of $S_1$ and $S$ is bivariate standard normal with correlation coefficient

$$\rho\left(S_1, S\right) = \sqrt{\left(m_1 + n_1\right)/\left(m+n\right)} \quad \text{(C-15)}$$

Some simulation studies would be needed to determine some quantitative bounds on the accuracy of this approximation.

## C.4 An Alternative Two-Stage, Two-Sample Median Test

A different approach to this testing problem has been suggested by Wolfe (1977). In this procedure, a specific number of sample measurements are made in a reference area, and the median, *M*, is calculated and the $DCGL_W$ added. Survey unit samples are then analyzed until *r* of them are found to be below *M*. The test statistic, $n_r$, is the number of survey unit samples that have been analyzed. Smaller values of $n_r$ indicate that the survey unit meets the release criterion. For Class 2 and Class 3 survey units, in particular, one would expect that $n_r = r$. In that case, the number of reference area measurements, *m*, and the value of *r* are chosen to meet the DQOs for the Type I error rate. In each survey unit, *r* samples are taken. If all are less than *M*, one rejects the null hypothesis that the survey unit exceeds the release criterion. If any one of them exceeds *M*, the null hypothesis will not be rejected. Thus, the total number of samples needed in each survey unit may be relatively small. In addition, as soon as one sample is measured above *M*, the result of the test is known. Thus, it may not be necessary to analyze every survey unit sample. Of course, the need to identify elevated areas may preclude the use of this method in some circumstances. However, the potential savings when the analytical costs are high may make this procedure attractive. As stated previously, if a licensee

is considering the use of approaches discussed in this appendix, contact with the NRC staff is strongly encouraged early in the planning process.

## C.5 References


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# APPENDIX D

# SURVEY DATA QUALITY AND REPORTING

## D.1 INTRODUCTION

The MARSSIM and the MARLAP manual are complementary guidance documents in support of cleanup and decommissioning activities. The MARSSIM document provides guidance on how to plan and carry out a study to demonstrate that a site meets appropriate release criteria. It describes a methodology for planning, conducting, evaluating, and documenting environmental radiation surveys conducted to demonstrate compliance with cleanup criteria. Chapter 4 and Appendix A provide more details on MARSSIM. The MARLAP manual provides guidance and a framework for both project planners and laboratory personnel to ensure that radioanalytical data will meet the needs and requirements of cleanup and decommissioning activities.

Radioanalytical data are commonly generated to support activities such as characterization and survey of radiologically contaminated sites, effluent and environmental monitoring of nuclear facilities, emergency response to accidents involving radiological materials, cleanup and decommissioning of nuclear facilities, and radioactive waste management. Numerous significant decisions, affecting the health and safety of the public and the environment, are frequently based on the available radioanalytical data. Considering these activities, the decisions associated with the radioanalytical data may involve issues pertaining to the extent and depth of contamination and associated remedial actions; demonstration of compliance with the cleanup criteria; demonstration of compliance with the effluent release criteria; assessment of effluent radiological releases and corrective measures; assessment of actions in response to incidents or accidental releases of radiological materials; and issues involving waste storage, transport, and disposal. In addition, radioanalytical data commonly influence decisions related to the cost of remedial actions as well as decisions involving environmental monitoring strategies and designs.

The MARLAP manual was developed to provide guidance and a framework for project planners, managers, technical reviewers, and laboratory personnel to ensure that the radioanalytical data produced by surveys will meet the needs and requirements for cleanup and decommissioning activities. The MARLAP manual addresses the need for a nationally consistent approach to producing radioanalytical laboratory data that meet a project's or program's data requirements. The guidance provided by MARLAP is both scientifically rigorous and flexible enough to be applied to a diversity of projects and programs. The MARLAP manual (NRC document NUREG-1576 and U.S. Environmental Protection Agency (EPA) document EPA 402-B-04-001A-C) is issued in three volumes (printed version and CD-ROM) and is found on the Internet at: http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1576.

The NRC staff encourages licensees to follow the recommendations in MARLAP.

## D.2 An Overview of MARLAP

The MARLAP manual is divided into two main parts. Part I provides guidance on using a performance-based approach for the three phases of radioanalytical projects, including (1) the planning phase, (2) the implementation phase, and (3) the assessment phase. These three main phases and associated processes should result in analytical data of known quality appropriate for the intended use. Table D.1 provides an overview of the three main phases, the processes associated with each phase, and the anticipated outputs for each process. Figure D.1 illustrates an overview of MARLAP terms and processes and interactions of the radioanalytical project manager with the laboratory performing the analysis. The MARLAP manual processes and terms described in Table D.1 and Figure D.1 are consistent with standard practices of the American Society for Testing of Materials (ASTM) for the generation of

environmental data. Chapters 3 through 9 of the MARLAP manual provide a detailed description of MARLAP phases and specific processes. It should be noted that it is not a regulatory requirement to follow or use MARLAP processes as described in Figure D.1; however, these processes are believed to be flexible and scientifically rigorous to be applied for the generation of radioanalytical data of the desired quality for the intended use.

Part II of MARLAP provides technical information on the laboratory analysis of radionuclides. Specifically, Part II highlights common radioanalytical problems and how to correct them. It also provides options for analytical protocols and discusses the pros and cons of these options. It should be noted that Part II does not provide step-by-step instructions on how to perform certain laboratory procedures or tasks. However, Part II provides guidance to assist laboratory personnel in selecting the best approach for a particular laboratory task. For example, Chapter 13 does not contain a step-by-step instruction on how to dissolve a soil sample; however, it does provide information on acid digestion, fusion techniques, and microwave digestion, to help the analyst select the most appropriate technique or approach for particular sample characteristics and project needs. Part II presents detailed technical information on (1) field and sampling issues that affect laboratory measurements, (2) sample receipt, inspection, and tracking, (3) laboratory sample preparation, (4) sample dissolution, (5) separation techniques, (6) quantification of radionuclides, (7) data acquisition, reduction, and reporting for nuclear counting instrumentation, (8) waste management in a radioanalytical laboratory, (9) laboratory quality control, (10) measurement uncertainty, and (11) detection and quantification capabilities. MARLAP adopted the International Organization for Standardization (ISO) processes, terms, and expressions for analytical measurements, quantifications, and estimation of uncertainty.

MARLAP also presents technical details on specific topics outlined in Parts I and II. Appendices A through E support Part I for the following specific topics: Appendix A, “Directed Planning Approaches”; Appendix B, “The Data Quality Objective Process”; Appendix C, “Measurement Quality Objectives for Method Uncertainty and Detection and Quantification Capability”; Appendix D, “Content of Project Plan Documents”; and Appendix E, “Contracting Laboratory Services”. Appendix F supports Part II for the specific topic on laboratory subsampling, and Appendix G provides a compilation of statistical tables.

## D.3 Use of MARLAP in Decommissioning and Cleanup Projects

MARLAP presents a useful approach and methodology applicable to radioanalytical projects for cleanup and decommissioning activities. The major processes of the Data Life Cycle are described briefly below for application in cleanup and decommissioning activities:

### D.3.1 The Planning Phase

As illustrated in Table D.1 and Figure D.2, planning documents could include Quality Assurance Project Plans (or QAPPs), Work Plans, Sampling and Analysis Plans, Data Validation Plans, and Data Quality Assessment Plans. Different organizations may use different terms for these documents but typically the set of documents include common elements. As provided in ANSI/ASQC E-4, QAPPs or other planning documents should detail the QA, QC, and other technical requirements that must be implemented to ensure that the results of the work will meet stated performance criteria. MARLAP selected EPA’s QAPP as a model for project plan documents because (i) it is closely associated with the DQO planning process, and (ii) widely accepted guidance on content (EPA, 2004; EPA, 2002). Chapter 4 and Appendix D of MARLAP contain additional information on the scope and content of planning documents.

The directed planning process for cleanup and decommissioning typically involves the following radioanalytical aspects:

- **Stating the cleanup problem:** Identify the analytes of concern, matrix of concern, regulatory requirements, sampling constraints, primary decisionmakers, available resources, and existing data and their reliability.

- **Identifying the cleanup decision:** Assess different analytical protocols, identify items of the analytical protocol specifications, and determine how sample collection will affect the measurement quality objectives (MQOs).

- **Identifying the inputs to the cleanup decisions:** Define characteristics of the analytes and matrix, assess the concentration range for the analyte of interest, and define action levels.

- **Defining the decision boundaries:** Identify background and temporal and spatial trends of data and determine limitations of current analytical protocols.

- **Developing a decision rule and tolerable decision error rates:** For example, the decision rule may be defined as, "If the mean concentration of analyte x in the upper 15 cm of the soil is greater than z Bq/g, then an action would be taken to remove the soil from the site." Estimates should be made of uncertainties in the data considering action levels and/or derived concentration guidelines.

- **Specifying limits on decision error rates:** Evaluate the range of possible parameter values and the allowable difference between the action level and the actual value.

- **Optimizing the strategy for obtaining data:** This process may involve optimization of the design for data collection through coordination with the different team members. The process also involves developing analytical protocols specifications and establishing performance measures of the MQOs.

### D.3.2 The Implementation Phase

The radioanalytical process is a compilation of activities, starting from the time a sample is collected and ending with the data reduction and reporting. Figure 2 illustrates the typical components of an analytical process used for radiological characterization and survey of contaminated sites. Certain cleanup or decommissioning projects may not include all of the components listed in Figure 2. The analytical protocols usually comprise a compilation of specific procedures or methods and are performed in succession, depending on the particular analytical process. Using a performance-based approach, a number of alternative protocols might be appropriate for a particular analytical process. A major component of the analytical protocol is the analytical method. The radioanalytical process should also include analytical uncertainty, analytical error, precision, bias, and accuracy of the method used.

### D.3.3 The Assessment Phase

The assessment phase focuses on three major steps:

(1) **Data verification:** This step ensures that the laboratory conditions and operations are in compliance with the statement of work and the project's QA project plan. The

verification process would examine the laboratory standard operating procedures. It would also check for consistency and comparability of the data, correctness of the data calculations, and completeness of the results and data documentation.

(2) **Data validation:** This step addresses the reliability of the radioanalytical data. It addresses the analyte and matrix types, as well as the uncertainty of the measurement to support the intended use. Validation flags (qualifiers) are typically applied to data that do not meet the acceptance criteria established to meet the project DQOs and MQOs.

(3) **Data quality assessment (DQA):** This step represents the scientific and statistical data evaluation aspects to determine if data are of the right type, quality, and quantity to support the intended use. The DQA is more global in its purview, such that it considers the combined impacts of all project activities on data quality and its usability.

## D.4 Benefits of Using MARLAP in Decommissioning and Cleanup Projects

MARLAP is an extensive document that presents comprehensive guidance and information on the three phases of the radioanalytical Data Life Cycle. MARLAP emphasizes the importance of establishing the proper linkages among these phases. Use of MARLAP in decommissioning and cleanup projects can benefit the user in the following respects:

- MARLAP ensures the generation of radioanalytical data of acceptable quality for the intended use.
- MARLAP minimizes time and effort expended in generating unacceptable data.
- MARLAP enhances public trust in the radioanalytical data generated by licensees and regulators.
- MARLAP minimizes efforts applied to justifying data and may limit any litigation costs.
- Because MARLAP uses an early coordinated approach to develop the radioanalytical data DQOs and MQOs, this approach would require early coordination and inputs from the decisionmakers, the project manager, shareholders, concerned team members, and the analyst (see Figure D.1). Therefore, this approach should resolve issues or difficulties related to sampling, sample tracking, sample preservation, analysis, data quality, time, and costs early in the process.
- MARLAP provides flexibility in selecting the appropriate analytical method, using a performance-based approach that considers the DQOs, the MQOs, and the available resources.
- MARLAP enhances regulatory reviews of radioanalytical data and saves time and effort for site characterization, environmental monitoring, decommissioning, and remediation.

**Table D.1 The Radioanalytical Data Life Cycle**

| PHASE | PROCESS | PROCESS OUTPUTS |
|---|---|---|
| PLANNING | Directed Planning Process | Development of DQOs and MQOs, including Optimized Sampling and Analytical Designs |
| | Plan Documents | Project Plan Documents, including Quality Assurance Project Plan (QAPP), Work Plan, or Sampling and Analysis Plan, Data Validation Plan, Data Quality Assessment Plan |
| | Contracting Services | Statement of Work and Other Contractual Documents |
| IMPLEMENTATION | Sampling | Laboratory Samples |
| | Analysis | Laboratory Analysis, including QC Samples and Complete Data Package |
| ASSESSMENT | Verification | Verified Data and Data Verification Report |
| | Validation | Validated Data and Data Validation Report |
| | DQA | Assessment Report |
| **Data of Known Quality Appropriate for the Intended Use** | | |

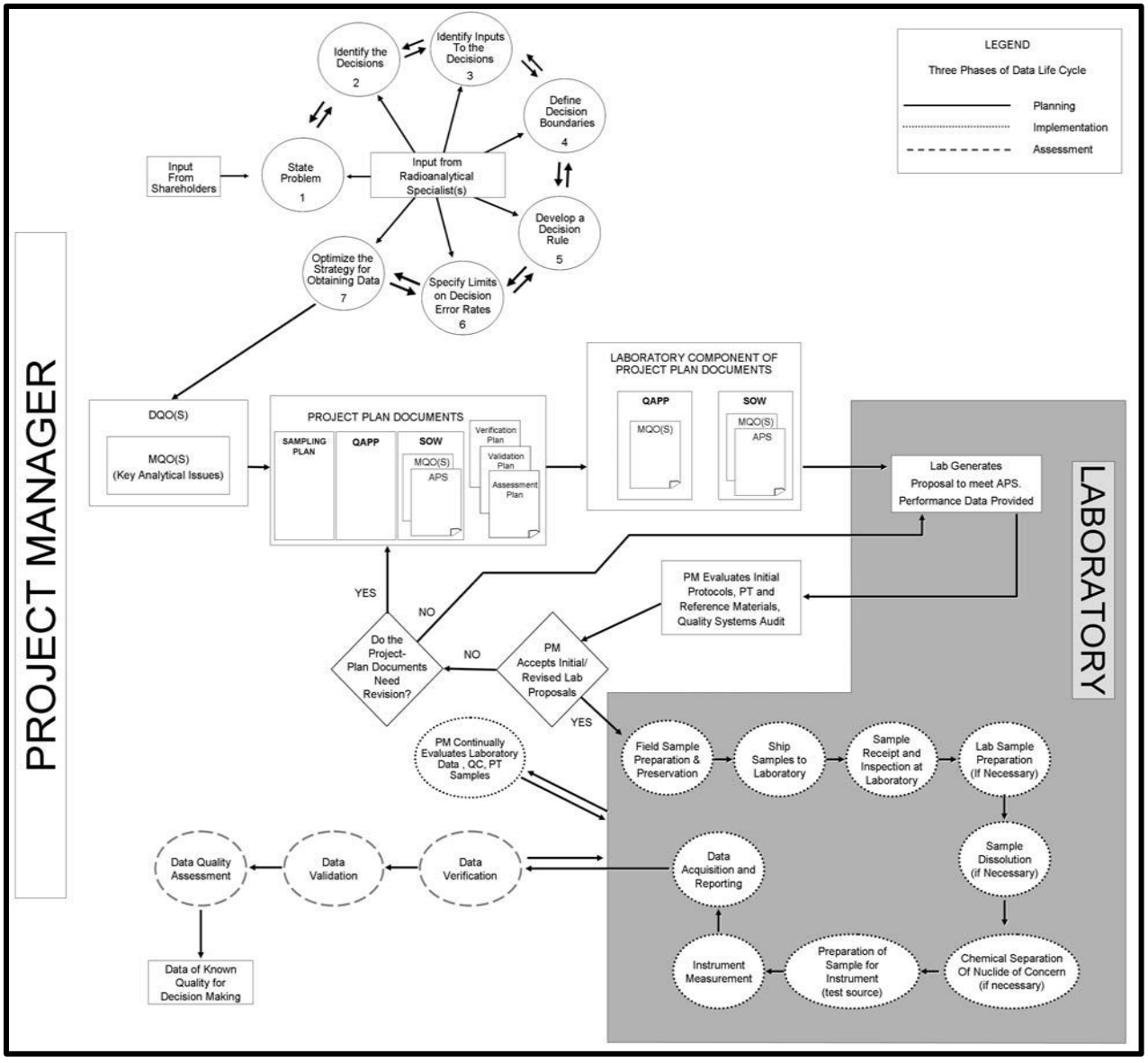


**Figure D.1 MARLAP Road Map—Key Terms and Processes**

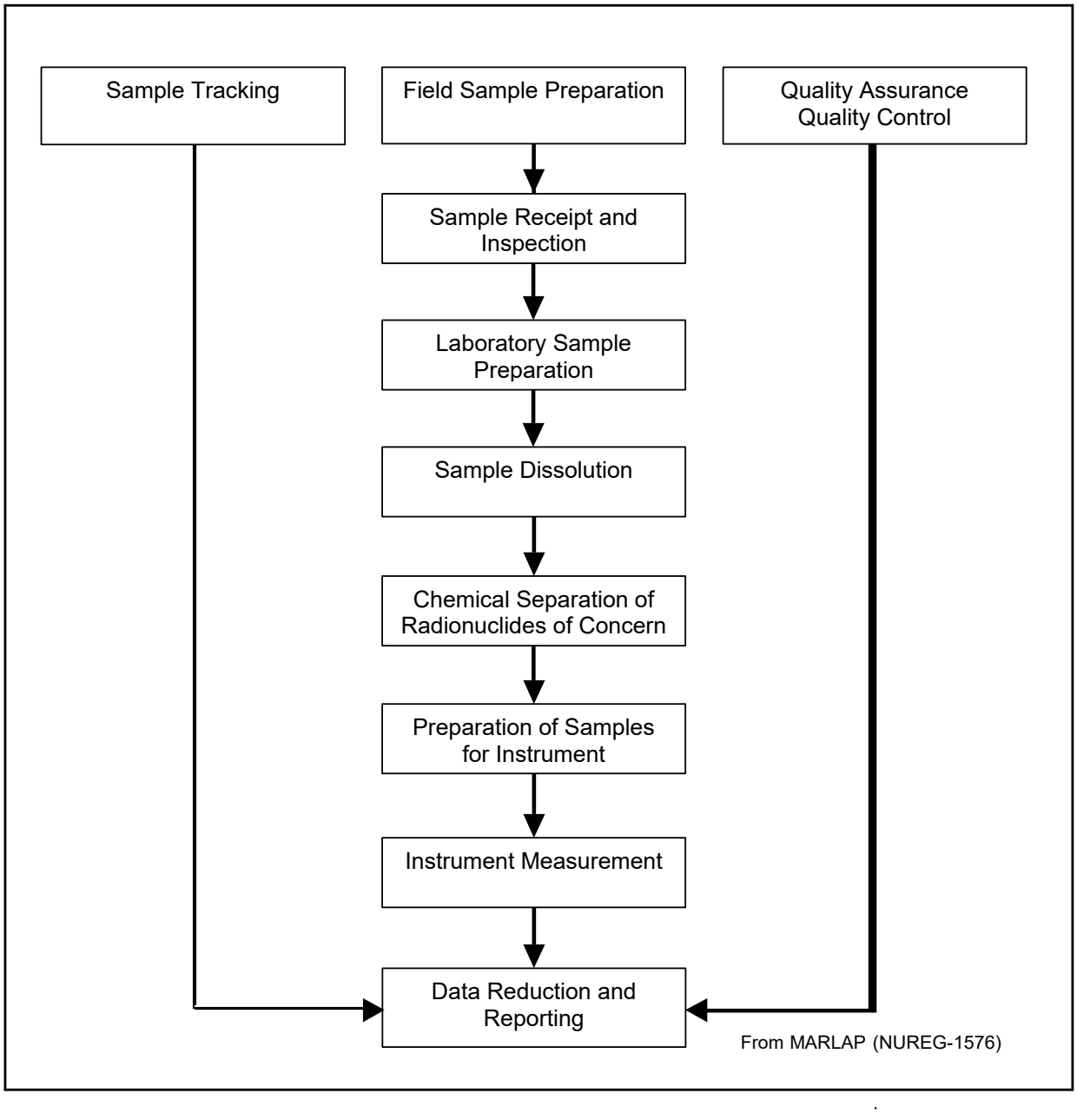


**Figure D.2 Typical Components of the Radioanalytical Process**

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# APPENDIX E

# MEASUREMENTS FOR FACILITY RADIATION SURVEYS

## E.1 Introduction

This appendix is applicable to all decommissioning groups. All surveys, whether simple or complex FSSs, require information on the reasons for instrument selection, the nature of the radionuclides, measurement techniques and procedures, MDCs of the instruments (measurement systems), and instrument calibration. Therefore, the information presented in this appendix would apply to a simple survey used to demonstrate compliance with regulatory decommissioning criteria, as well as a complex FSS.

This appendix contains limited, general information on survey techniques and survey measurements. The information presented here is related to the process of implementing a survey plan and refers to the appropriate sections of the MARSSIM (NRC, 2000)[1], MARLAP, and various NUREGs for more detailed information. These are important areas for the conduct of surveys in the RSSI process and include the basic modes for determining levels of radiation and radioactivity at a site, instrument and scanning detection limits, instrument calibration, and laboratory measurements for samples. The data from the FSS are the deciding factor in judging if the site meets the release criteria.

Radiological conditions that should be determined for license termination purposes include any combination of total surface activities, removable surface activities, exposure rates, radionuclide concentrations in soil, or induced activity levels. To determine these conditions, field measurements and laboratory analyses may be necessary. For certain radionuclides or radionuclide mixtures, the licensee may have to measure both alpha and beta radiation. In addition to assessing the average radiological conditions, the licensee should identify small areas with elevated levels of residual radioactivity and determine their extent and activities. There are three basic modes in which one can operate in determining the levels of radiation and radioactivity at a site. They are scanning with hand-held survey instruments, direct measurements with these same or larger instruments, and sample collection at the site followed by analysis in the laboratory. In many cases, the licensee will use some combination of these modes to obtain data, although the exact mix would be expected to vary according to the application.

In practice, the licensee uses the DQO process to obtain a proper balance among the uses of various measurement techniques. In general, there is an inverse correlation between the cost of a specific measurement technique and the detection levels being sought. Depending on the survey objectives, important considerations include survey costs and choosing the optimum instrumentation and measurement mix.

The decision to use a measurement method as part of the survey design is determined by the survey objectives and the survey unit classification. Scanning is performed to identify areas of elevated activity that other measurement methods may not detect. Direct measurements are analogous to collecting and analyzing samples to determine the average activity in a survey unit.

[1] It is important to note that as of the date of publication of this draft volume, MARSSIM, Revision 1, is the current version of MARSSIM, while MARSSIM, Revision 2, is currently being drafted. Any changes to MARSSIM guidance in the future that affects the guidance in this volume will be reflected in future revisions to this volume. The decommissioning website should be consulted for issuance of interim guidance and lessons learned between guidance revisions.

## E.2 Direct Measurements (Fixed Measurements)

To conduct direct measurements of alpha, beta, and photon surface activity, instruments and techniques providing the required detection sensitivity are selected. The selection of the type of instrument and method of performing the direct measurement depends on the type of residual radioactivity present, the measurement sensitivity requirements, and the objectives of the radiological survey.

Direct measurements may be collected at random locations in the survey unit. Alternatively, direct measurements may be collected at systematic locations and supplement scanning surveys to identify small areas of elevated activity. Direct measurements may also be collected at locations identified by scanning surveys as part of an investigation to determine the source of the elevated instrument response. Professional judgment may also be used to identify locations for direct measurements to further define the areal extent of residual radioactivity and to determine maximum radiation levels within an area, although these types of direct measurements are usually associated with preliminary surveys (i.e., scoping, characterization, remedial action support). Licensees should document all direct measurement locations and results.

If the equipment and methodology used for scanning are capable of providing data of the same quality required for direct measurement *(*e.g., detection limit, location of measurements, ability to record and document results), then scanning may be used in place of direct measurements. Similarly, the usage of *in situ* measurement instrumentation may be possible if sufficient data quality can be achieved. In both cases, proposed approaches should be developed using the DQO process and should be communicated to NRC staff. Results should be documented for at least the number of locations required for the statistical tests. In addition, some direct measurement systems may be able to provide scanning data, provided they meet the objectives of the scanning survey.

Chapter 6 of MARSSIM includes information on radiation measurements. Specifically, Section 6.4.1 of MARSSIM contains information on direct measurements for alpha-, beta-, and gamma-emitting radionuclides.

## E.3 Scanning Measurements

Scanning is the process by which the operator uses portable radiation detection instruments to detect the presence of radionuclides on a specific surface (i.e., ground, wall, floor, equipment). The term scanning survey describes the process of moving portable radiation detectors across a suspect surface with the intent of locating residual radioactivity. Investigation levels for scanning surveys are determined during survey planning to identify areas of elevated activity. Scanning surveys are useful in locating radiation anomalies indicating residual gross activity that might require further investigation or action.

Areas of elevated activity typically represent a small portion of the site or survey unit. Thus, random or systematic direct measurements or sampling on the commonly used grid spacing may have a low probability of identifying these areas. Scanning surveys are often relatively quick and inexpensive to perform. For these reasons, the licensee typically performs them before direct measurements or sampling. This avoids spending time fully evaluating an area that may quickly prove to contain residual radioactivity above the IL during the scanning process. Based on the HSA, surfaces to be surveyed, and survey design objectives, licensees conduct scans that would indicate all radionuclides potentially present, using surrogate

measurements where appropriate. Documenting scanning results and observations from the field is very important. For example, licensees should document a scan that identified relatively sharp increases in instrument response or identified the boundary of an area of increased instrument response. This information is useful when interpreting survey results.

Chapter 6 of MARSSIM includes information on radiation measurements. Specifically, Section 6.4.2 of MARSSIM contains information on scanning measurements for alpha-, beta-, and gamma-emitting radionuclides.

## E.4 Sampling

For certain radionuclides that cannot be effectively measured directly in the field, the licensee should collect samples of the medium under investigation (e.g., soil) and then analyze them with a laboratory-based procedure. On the simplest level, this would include the analysis of a smear sample using a gross alpha-beta counter. More involved analyses would include gamma spectrometry, beta analysis using liquid scintillation counting, or alpha spectrometry following separation chemistry.

Samples from a variety of locations may be required, depending upon the specific facility conditions and the results of scans and direct measurements. Inaccessible surfaces cannot be adequately evaluated by direct measurements on external surfaces alone; therefore, those locations that could contain residual radioactive material should be accessed for surveying. Residue can be collected from drains using a piece of wire or plumber's "snake" with a strip of cloth attached to the end; deposits on the pipe interior can be loosened by scraping with a hard-tipped tool that can be inserted into the drain opening. Particular attention is paid to "low points" or "traps" where activity would likely accumulate. The need for further internal monitoring and sampling is determined on the basis of residue samples and direct measurements at the inlet, outlet, cleanouts, and other access points to the pipe interior.

Residual activity will often accumulate in cracks and joints in the floor. These are sampled by scraping the crack or joint with a pointed tool such as a screwdriver or chisel. Samples of the residue can then be analyzed; positive results of such an analysis may indicate possible subfloor residual radioactivity. Checking for activity below the floor will require accessing a crawl space (if one is present) or removing a section of the flooring. Coring, using a commercially available unit, is a common approach to accessing the subfloor soil. After removing the core (where the diameter may range from a few centimeters to up to 20 centimeters), direct monitoring of the underlying surface can be performed and samples of soil collected.

Coring is also useful for collecting samples of construction material that may contain activity that has penetrated below the surface or activity induced by neutron activation. This type of sampling is also applicable to roofing material, which may contain embedded or entrapped contaminants. The profile of the distribution and the total radionuclide content can be determined by analyzing horizontal sections of the core.

If residual activity has been coated by paint or some other treatment, the underlying surface and the coating itself may contain residual radioactivity. If the activity is a pure alpha or low-energy beta emitter, measurements at the surface will probably not be representative of the actual residual activity level. In this case, the licensee can remove the surface layer from a known area, usually 100 $cm^2$, using a commercial stripping agent or by physically abrading the surface. The removed coating material is analyzed for activity content and the level converted to units of

disintegrations per minute (dpm)/100 square centimeters ($cm^2$) for comparison with guidelines for surface activity. The licensee takes direct measurements on the underlying surface, after removing the coating.

MARSSIM and MARLAP contain information on sampling and laboratory analysis for decommissioning. Chapter 10 of MARLAP discusses field and sampling issues that affect laboratory measurements.

## E.5 Minimum Detectable Concentrations

Detection limits for field survey instrumentation are important criteria in the selection of appropriate instrumentation and measurement procedures. For the most part, the licensee determines detection limits to evaluate whether a particular instrument and measurement procedure is capable of detecting residual activity at the regulatory release criteria. One may demonstrate compliance with decommissioning criteria by performing surface activity measurements and directly comparing the results to the surface activity DCGLs. However, before any measurements are performed, the survey instrument and measurement procedures to be used should be evaluated to ensure they possess sufficient detection capabilities relative to the surface activity DCGLs.

The measurement of residual radioactivity during surveys in support of decommissioning often involves measuring residual radioactivity at near-background levels. Thus, the licensee should determine the minimum amount of radioactivity that may be detected by a given survey instrument and measurement procedure. In general, the MDC is the minimum activity concentration on a surface or within a material volume that an instrument is expected to detect (i.e., activity expected to be detected with 95 percent confidence). It is important that this activity concentration, the MDC, is determined *a priori* (i.e., before survey measurements are conducted).

As generally defined, the detection limit, which may be a count or count rate, is independent of field conditions such as scabbled, wet, or dusty surfaces. That is, the detection limit is based on the number of counts and does not necessarily equate to measured activity under field conditions. These field conditions do, however, affect the instrument's "detection sensitivity" or MDC. Therefore, the licensee should not use the terms MDC and detection limit interchangeably.

In MARSSIM, MARLAP, and other NRC NUREGs, the MDC corresponds to the smallest activity concentration measurement that is practically achievable with a given instrument and type of measurement procedure. That is, the MDC depends not only on the particular instrument characteristics (e.g., instrument efficiency, background, integration time) but also on the factors involved in the survey measurement process (U.S. Environmental Protection Agency, EPA 520/1-80-012, "Upgrading Environmental Radiation Data," issued August 1980), which include surface type, source-to-detector geometry, and source efficiency (e.g., backscatter and self-absorption).

MARLAP Section 3.3.7, "Method Performance Characteristics and Measurement Quality Objectives," and Chapter 20, "Detection and Quantification Capabilities," discuss MDCs.

## E.6 Survey Minimum Detectable Concentrations

During radiological surveys in support of decommissioning, scanning is performed to identify the presence of any locations of elevated direct radiation. The probability of detecting residual radioactivity in the field is affected not only by the sensitivity of the survey instrumentation when used in the scanning mode of operation but also by the surveyor's ability. The surveyor will decide whether the signals represent only the background activity, or whether they represent residual radioactivity in excess of background.

The MDC of a scan survey, referred to as scan MDC or $MDC_{scan}$, depends on the intrinsic characteristics of the detector (e.g., efficiency, window area), the nature (e.g., type and energy of emissions) and the relative distribution of the residual radioactivity (e.g., point versus distributed source and depth of residual radioactivity), scan rate, and other characteristics of the surveyor. Some factors that may affect the surveyor's performance include the costs associated with various outcomes—e.g., cost of missed residual radioactivity versus cost of incorrectly identifying areas as containing residual radioactivity—and the surveyor's *a priori* expectation of the likelihood of residual radioactivity being present. For example, if the surveyor believes that the potential for residual radioactivity is very low, as in an unaffected area, then a relatively large signal may be needed for the surveyor to conclude that residual radioactivity is present. NUREG/CR–6364, "Human Performance in Radiological Survey Scanning," issued March 1998, contains a complete discussion of the human factors as they relate to the performance of scan surveys.

Signal detection theory provides a framework for the task of deciding whether the audible output of the survey meter during scanning was due to background or signal plus background levels. An index of sensitivity ($d'$) that represents the distance between the means of the background and background plus signal, in units of their common standard deviation, can be calculated for various decision errors—Type I error (α) and Type II error (β). As an example, for a correct detection or true positive rate of 95 percent (1−β) and a false positive rate (α) of 5 percent, $d'$ is 3.29 (similar to the static MDC for the same decision error rates). The index of sensitivity is independent of human factors, and therefore, the ability of an ideal observer (i.e., theoretical construct) may be used to determine the minimum $d'$ that can be achieved for particular decision errors. The ideal observer makes optimal use of the available information to maximize the percent of correct responses and thus provides an effective upper bound against which to compare actual surveyors. Computer simulations and field experimentation can then be performed to evaluate the surveyor efficiency ($p$) relative to the ideal observer. The resulting expression for the ideal observer's minimum detectable count rate ($MDCR$), in counts per minute (cpm), can be written:

$$MDCR = d' \times \sqrt{b_i} \times (60 / i) = s_i \times (60 / i) \quad \text{(E-1)}$$

where $MDCR$ = minimum detectable (net) count rate in cpm,
$b_i$ = background counts in the observation interval,
$s_i$ = minimum detectable number of net source counts in the observation interval, and
$i$ = observational interval (in seconds), based on the scan speed and areal extent of the residual radioactivity.

*Scan MDC*s are determined from the *MDCR* by applying conversion factors to obtain results in terms of measurable surface activities and soil concentrations. As an example, the *scan MDC* for a structure surface can be expressed as:

$$Scan\ MDC = \frac{MDCR}{\sqrt{p}\ \varepsilon_i\ \varepsilon_s \frac{probe\ area}{100\ cm^2}} \qquad \text{(E-2)}$$

Chapter 6 of NUREG-1507, "Minimum Detectable Concentrations with Typical Radiation Survey Instruments for Various Contaminants and Field Conditions," issued June 1998, contains a discussion of survey MDCs. This discussion includes *scan MDC* equations for both building/structure surfaces and land areas.

## E.7 Survey Instrument Calibration

Before determining the MDC for a particular instrument and survey procedure, it is necessary to introduce the expression for total alpha or beta surface activity per unit area. In the ISO Guide 7503–1, "Evaluation of Surface Contamination," 1988, the ISO recommends calculating the total surface activity, $A_S$, as in the following expression:

$$\mathrm{A}_s = \frac{R_{S+B} - R_B}{(\varepsilon_i)(W)(\varepsilon_S)} \qquad \text{(E-3)}$$

where $R_{S+B}$ = the gross count rate of the measurement in cpm,
$R_B$ = the background count rate in cpm,
$\varepsilon_i$ = the instrument or detector efficiency (unitless),
$\varepsilon_s$ = the efficiency of the residual radioactivity source (unitless), and
$W$ = the area of the detector window ($cm^2$).

(For instances in which $W$ does not equal 100 $cm^2$, probe area corrections are necessary to convert the detector response to units of dpm per 100 $cm^2$.)

This expression clearly distinguishes between instrument (detector) efficiency and source efficiency. The product of the instrument and source efficiency yields the total efficiency, $\varepsilon_{tot}$. Currently, surface residual radioactivity is assessed by converting the instrument response to surface activity using one overall total efficiency. This is not a problem, provided that the calibration source exhibits characteristics similar to the surface residual radioactivity—including such characteristics as radiation energy, backscatter effects, source geometry, and self-absorption. In practice, this is hardly the case; more likely, total efficiencies are determined with a clean, stainless steel source, and then those efficiencies are used to measure residual radioactivity on a dust-covered concrete surface. By separating the efficiency into two components, the surveyor has a greater ability to consider the actual characteristics of the surface residual radioactivity.

The instrument efficiency is defined as the ratio between the net count rate of the instrument and the surface emission rate of a source for a specified geometry. The surface emission rate, $q_{2\pi}$, is defined as the "number of particles of a given type above a given energy emerging from

the front face of the source per unit time" (ISO 7503–1). The surface emission rate is the 2π particle fluence that embodies both the absorption and scattering processes that affect the radiation emitted from the source. Thus, the instrument efficiency is determined by

$$A_S = \frac{R_{S+B} - R_B}{(\varepsilon_i)(W)(\varepsilon_S)} \quad \text{(E-4)}$$

The instrument efficiency is determined during calibration by obtaining a static count with the detector over a calibration source that has a traceable activity or surface emission rate or both. In many cases, it is the source surface emission rate that is measured by the manufacturer and certified as National Institute of Standards and Technology traceable. The source activity is then calculated from the surface emission rate, based on the assumed backscatter and self-absorption properties of the source. The theoretical maximum value of instrument efficiency is one.

The source efficiency, $\varepsilon_s$, is defined as the ratio between 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 (ISO 7503–1). The source (or surface) efficiency takes into account the increased particle emission due to backscatter effects, as well as the decreased particle emission due to self-absorption losses. For an ideal source (no backscatter or self-absorption), the value of $\varepsilon_s$ is 0.5. Many real sources will exhibit values of $\varepsilon_s$ less than 0.5, although values greater than 0.5 are possible, depending on the relative importance of the absorption and backscatter processes. Source efficiencies may either be determined experimentally or simply selected from the guidance contained in ISO 7503–1.

Some of the factors that affect the instrument efficiency, $\varepsilon_I$, include detector size (probe surface area), window density thickness, geotropism, instrument response time, and ambient conditions such as temperature, pressure, and humidity. The instrument efficiency also depends on the radionuclide source used for calibration and the solid angle effects, which include source-to-detector distance and source geometry.

Some of the factors that affect the source efficiency, $\varepsilon_s$, include the type of radiation and its energy, source uniformity, surface roughness and coverings, and surface composition (e.g., wood, metal, concrete).

The licensee assesses surface activity levels by converting detector response, through the use of a calibration factor, to radioactivity. Once the detector has been calibrated and an instrument efficiency ($\varepsilon_i$) established, several factors still need to be carefully considered when using that instrument in the field. These factors involve the background count rate for the particular surface and the surface efficiency ($\varepsilon_s$), which addresses the physical composition of the surface and any surface coatings. Ideally, the surveyor should use experimentally determined surface efficiencies for the anticipated field conditions. The surveyor needs to know how and to what degree these different field conditions can affect the sensitivity of the instrument. A particular field condition may significantly affect the usefulness of a particular instrument (e.g., wet surfaces for alpha measurements or scabbled surfaces for low-energy beta measurements).

One of the more significant implicit assumptions commonly made during instrument calibration and subsequent use of the instrument in the field is that the composition and geometry of residual radioactivity in the field is the same as that of the calibration source. This may not be the case, considering that many calibration sources are fabricated from materials different from those that comprise the surfaces of interest in the field (e.g., activity plated on a metallic disc (Walker, "Proper Selection and Application of Portable Survey Instruments for Unrestricted Release Surveys," issued 1994)). This difference usually manifests itself in the varying backscatter characteristics of the calibration and field surface materials.

Generally, it will not be necessary to recalculate the instrument MDC to adjust for the field conditions. The instrument detection limit (in net counts or net count rate) remains the same, but the surface activity MDC may be different (due to the varying $\varepsilon_s$).

It is important to note that the preceding discussion on source efficiency and the calculation of surface activity are based on guidance in ISO 7503-1:1988[2]. The ISO 7503 series was updated in 2016, prior to Revision 2 of this NUREG report. Therefore, a comparison of the 1988 and 2016 versions of ISO 7503 was performed to understand any differences in approach to surface or source efficiency, and to determine if the methodology/terminology in NUREG-1507 should be updated. A detailed discussion of this analysis will be provided in Revision 1 of NUREG-1507[3], while a brief overview of the comparison is provided below.

Several of the basic concepts and methods were compared between the 1988 and 2016 ISO 7503 series. The concepts of "source efficiency" and an "ideal source" were compared, and it was concluded that they were presented as essentially the same (noting that a different "*P*-Factor" terminology was used for the 2016 series). The equations to evaluate contamination measurement data, and the associated usage of $\varepsilon_s$ and the *P*-Factor, were compared between the two series, and these equations were essentially the same (while different terminology has been used). A comparison was performed between recommended default or "conservative" $\varepsilon_s$ values and *P*-Factors in the two series, and these values utilized the same assumptions and made the same recommendations for default source efficiencies (again noting that different terminology was presented). Based upon the ISO 7503 series comparison, it has been concluded that the 2016 revision of ISO 7503-1 presented no compelling reason to update the usage of the "surface efficiency" concept and terminology for Revision 1 of NUREG-1507. Therefore, the equations for MDC and surface activity measurement, which were originally developed using ISO 7503-1:1988 remain valid and are not changed from the previous NUREG-1757 version.

However, the comparison between the 1988 and 2016 ISO 7503 series identified that there is a need to consider weighted detection efficiencies for use with multiple radionuclides or with complex decay series, as ISO 7503-3:2016, in particular, presented many new concepts in this area. To address this need, Revision 1 to NUREG-1507 will include "weighted efficiency" calculations that utilize the concepts of instrument efficiency, source efficiency, and emission intensity, while also considering the relative fraction of radionuclides and branching ratios.

Chapter 4 of NUREG-1507 covers survey instrument calibration and the effects of efficiency changes on MDC. Chapter 5 of NUREG-1507 discusses variables affecting efficiencies in the

2 MARSSIM, Rev. 1, and NUREG-1507, Rev. 0, reference ISO 7503-1:1988. The NRC has found this standard acceptable for use by NRC licensees to calculate MDCs and surface activity measurements.

3 As of the date of publication of this volume, NUREG-1507, Revision 1, is under development.

field. Chapter 20 of MARLAP discusses instrument efficiency and the minimum detectable net instrument signal.

## E.8 Laboratory Measurements

Frequently during surveys in support of decommissioning, it is not feasible, or even possible, to detect the residual radioactivity with portable field instrumentation; thus arises the need for a laboratory analysis of media samples. This is especially the case for such media samples as soil, which result in significant self-absorption of the radiation from the residual radioactivity. Another common situation that necessitates the use of laboratory analyses occurs when the residual radioactivity is difficult to detect even under ideal conditions. This includes residual radioactivity that emits only low-energy beta radiation (e.g., H-3 and Ni-63) or x-ray radiation (e.g., Fe-55). Laboratory analyses for radionuclide identification, using spectrometric techniques, are often performed during scoping or characterization surveys. Here, the principal objective is to simply determine the specific radionuclides present in the residual radioactivity, without necessarily having to assess the quantity of residual radioactivity. Once the licensee identifies the residual radioactivity, it selects sufficiently sensitive field survey instrumentation and techniques to demonstrate compliance with the DCGLs.

Samples collected during surveys for decommissioning purposes should be analyzed by trained individuals using the appropriate equipment and procedures at a well-established laboratory, which uses either in-house or contractor laboratory services. There should be written procedures that document both (1) the laboratory's analytical capabilities for the radionuclides of interest and (2) the QA/QC program that ensures the validity of the analytical results. Many of the general types of radiation detection measuring equipment used for survey field applications are also used for laboratory analyses, usually under more controlled conditions that provide for lower detection limits and greater delineation between radionuclides. Laboratory methods often also involve a combination of both chemical and instrumental techniques to quantify the low levels expected to be present in samples from decommissioning facilities.

To reemphasize, a thorough knowledge of the radionuclides present, along with their chemical and physical forms and their relative abundance, is needed to select appropriate laboratory methods. With this information, it may be possible to substitute certain gross (i.e., nonradionuclide-specific) measurement techniques for the more costly and time-consuming wet chemistry separation procedures and relate the gross data back to the relative quantities of specific contaminants. The individual responsible for the survey should be aware that radiochemical analyses require lead times that will vary according to the nature and complexity of the request. For example, a laboratory could provide a fairly quick turnaround on gamma spectrometry because sample preparation is likely limited to the media being dried and homogenized before being measured in a standard geometry. In comparison, alpha spectrometry usually involves sample preparation that also includes chemical separation and will typically require a longer lead time. Some factors influencing the analysis time include (1) the nuclides of concern, (2) the type of samples to be analyzed, (3) the QA/QC considerations required, (4) the availability of adequate equipment and personnel, and (5) the required detection limits.

For relatively simple analyses, such as gross alpha and gross beta counting of smears and water samples, liquid scintillation spectrometry for low-energy beta emitters in smear and water samples, and gamma-spectrometry of soil, it is usually practical to establish in-house laboratory capabilities. The more complicated and labor-intensive procedures, such as alpha

spectrometry, Sr-90 and low-energy beta emitters (e.g., H-3, Ni-63) in soil samples, should be considered candidates for contract laboratory analyses.

Analytical methods should be capable of measuring levels below the established release guidelines; detection sensitivities of 10 to 25 percent of the guideline should be the target. Although laboratories will state detection limits, these limits are usually based on ideal situations and may not be achievable under actual measurement conditions. Also, remember that detection limits are subject to variation from sample to sample, instrument to instrument, and procedure to procedure, depending upon sample size, geometry, background, instrument efficiency, chemical recovery, abundance of the radiations being measured, counting time, self-absorption in the prepared sample, and interference from other radionuclides present.

MARSSIM and MARLAP contain information on sampling and laboratory analysis for decommissioning. MARLAP Sections 12, 13, 14, and 15 discuss laboratory sample preparation, sample dissolution, separation techniques, and quantification of radionuclides.

## E.9 References


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# APPENDIX F

# SURFACE WATER AND GROUNDWATER CHARACTERIZATION

## F.1 Introduction

This appendix includes guidance on surface and groundwater characterization to support development of conceptual site models, development of hydrologic inputs to dose assessment models, and radiological surveys to estimate dose associated with existing groundwater contamination. Surface water and groundwater characterization is important for Decommissioning Group 4 sites with surface water contamination and Decommissioning Groups 5–7 that have existing groundwater[1] contamination. Characterization is also important for sites that have a medium to high potential for groundwater contamination (see Section F.4) based on historical site activities even in the absence of evidence of groundwater contamination.

If groundwater is contaminated, characterization of groundwater is an essential component of the dose modeling used in the estimation of doses to demonstrate compliance with the license termination requirements in 10 CFR Part 20, "Standards for Protection against Radiation," Subpart E, "Radiological Criteria for License Termination." In these cases, unmodified screening derived concentration guideline levels (DCGLs) for soil are inappropriate to use, because the screening levels assume surface water and groundwater are initially uncontaminated. Likewise, if site-specific dose modeling is used to demonstrate compliance with radiological criteria for license termination, the dose contributions of contaminated surface water or groundwater should be taken into consideration in the dose assessment.

This appendix references reports, methods and software that should assist in characterization of surface water and groundwater and associated dose modeling. However, it should be noted that other reports, methods, and software may also be appropriate. If other approaches are utilized, it may be prudent for the licensee to contact NRC to ensure that the approaches are appropriate prior to any significant resource input by the licensee.

## F.2 Planning for Surface Water and Groundwater Characterization

The licensee should plan surface water and groundwater characterization in a manner that maximizes the utility of the information to be collected during the various stages of radiological surveys. For example, for a particular site, a licensee may show that the surface water pathway is not likely to be significant in terms of existing and potential future exposure to the public. In such a case, the need for detailed characterization of the surface water system decreases. On the other hand, the identification of groundwater contamination during the preliminary scoping survey may warrant installing and sampling additional monitoring wells to define the nature and extent of groundwater contamination.

In some instances, groundwater may be unsuitable for specific uses, such as human and livestock consumption, but may be acceptable for crop irrigation. In addition, some aquifers may not have the yield to support crop irrigation but may produce enough water for human consumption. In some instances, the EPA or a State agency may have declared that the aquifer in question is unfit for human or livestock use. Accordingly, the licensee should address this type of information, because it will support site exposure scenario development and dose modeling. The State agency may also have rules that apply to groundwater resource

[1] For the purposes of this volume, groundwater refers to water below the land surface in a zone of saturation, which can theoretically be used for drinking water and irrigation although arguments can be presented to eliminate these pathways of exposure (e.g., insufficient yield, water quality).

classifications. Section I.3.3.3 from Appendix I and Appendix M to this volume provides guidance on modification of waterborne exposure pathways.

## F.3 Development of CSMs and Mathematical Models

A conceptual site model (CSM) provides a hypothetical framework for contaminant source, geologic, hydrologic (including water usage), chemical, biologic, and demographic characteristics for the site (NRC, 2007). The CSM provides the basis for understanding flow and transport at the site for abstraction into a dose assessment model and is the starting point for numerical models if contamination is present in the surface water and groundwater. In general, CSMs should be updated as new information becomes available. The complexity of CSMs should be commensurate with site risk and at an appropriate level to demonstrate radiological criteria for license termination can be met. ASTM E1689-95 (2014) provides information on the development of CSMs that may be useful to licensees.

The model abstraction process involves the representation of major components of a complex system in a conceptual model. Conceptual models should represent or be able to describe the important features and processes of the system. A simplified representation of the conceptual model is developed so that the conceptual model can be more easily represented by a mathematical model. Mathematical models translate the assumptions of a conceptual model into the formalism of mathematics (IAEA, 2004). Mathematical models are represented by equations and, if these equations are solved, the output can include information such as radionuclide concentrations in media, doses to humans, and temporal evolution of the system. Numerical codes can solve such equations, and constructed numerical models are used to solve the equations represented in the mathematical models, thereby simulating the properties of features and processes represented in a conceptual model. Model simplification is the process for reducing the complexity of a complex numerical model into a simpler numerical model while still maintaining the validity of the simulation results and is another form of model abstraction.

Compared to model simplification (i.e. reducing a complex numerical model to a less complex numerical model), abstracting the physical reality that is represented in a conceptual model to a mathematical model is typically more vulnerable to error. In addition, it can be difficult to validate how successfully the physical system is abstracted into a mathematical model whereas it is usually easier to make sure that a simplified numerical model is adequately representing a more complex numerical model. However, unlike model simplification, model abstraction is a required and necessary step in the performance assessment methodology. Not all features and processes can or should be included in a mathematical model. As an analogy, news reporters must abstract facts from a real set of complex circumstances or else the news story becomes too lengthy to be useful for most readers or viewers. Not only is model abstraction necessary, it can be useful to an analyst in explaining complex system behavior by reducing the system to its major components and thereby improve analyst communications with stakeholders regarding the system being simulated. Additional information on model abstraction and simplification can be found in NUREG-7026 and NUREG-6884.

Model abstraction builds on insights gained from characterizing the system and developing exposure scenarios. In a dose assessment, several model abstractions typically support an overall assessment of a facility's ability to demonstrate compliance with release criteria. These abstractions usually include models of projected climate and infiltration, source term release, transport through environmental media including the groundwater, and potential exposures to a receptor in the biosphere. If surface water or groundwater are viable pathways, a CSM of the

hydrologic system is needed to adequately model these systems. Thus, adequate characterization of the subsurface is necessary to construct defensible hydrogeological conceptual models to perform dose modeling to estimate future impacts to surface water and groundwater, as well as to estimate doses from existing groundwater contamination. Although monitoring data can be useful in determining whether groundwater is contaminated and to determine the potential risk associated with groundwater dependent pathways, deficiencies in the monitoring well network and monitoring approach may compromise the ability to use existing monitoring well data for this purpose. In these cases, the licensee can manage uncertainty in exposure point concentrations through collection of additional data, and through additional modeling, including uncertainty/sensitivity analysis. Licensees should discuss acceptable approaches to assessing the risk and dose contributions associated with surface water and groundwater dependent pathways.

Environmental impact assessments for decommissioning may also benefit from the development of a CSM. The information needs of environmental impact assessments may differ slightly from a dose assessment for a safety case as non-radiological and other impacts associated with surface water and groundwater use must also be considered. Although specific to information needs to support preparation of environmental impact assessments, Regulatory Guide 4.2, Section 2.2, contains useful information regarding the collection of hydrological data for the purpose of environmental assessment, but may also be useful for creation of CSMs and development of dose assessment models (NRC, 2018).

It is also important to note that transients and spatial and temporal variability may need to be considered when simulating contaminant flow and transport if important to decision-making. For example, river stage may control groundwater flow direction under a site with changes in flow direction occurring over smaller timeframes, but net flow may be towards the river considering longer timeframes. A determination will need to be made if representation of short-term behavior is important to demonstrating compliance or will help provide a better understanding of system behavior and potential uncertainties in assessment of risk. Additional information on consideration of scale affects is provided in Appendix Q.

## F.4 Indicators for Potential Groundwater Contamination

The level of effort associated with subsurface characterizations is dependent in large part on the extent of residual radioactivity in the subsurface and the transport of residual radioactivity to groundwater. As described in Table 1.1 of Volume 1 of this NUREG report, Decommissioning Groups 5–7 are sites that have the potential for residual radioactivity in groundwater. Based on the experience gained from operational and decommissioning NRC licensed sites, the following is a list of potential indicators for groundwater contamination at decommissioning sites (NUREG-1496, Appendix C, Attachment E, Table C.E.1).[2] They are illustrative only and are not intended to constitute a complete list:

- High Potential: if a site has a history of, or currently has the following:
    - unlined lagoons, pits, canals, or surface-drainage ways that received radioactively contaminated liquid effluent

[2] MARSSIM, Rev. 1 (Sections 3.6.3.4 and 5.3.3.3) also provides guidance on evaluating the likelihood for release of residual radioactivity to groundwater, as well as characterization and sampling of groundwater (NRC, 2000).

- lined lagoons, pits, canals, or surface drainage ways that received radioactively contaminated liquid effluent, where the lining has leaked or ruptured, or where overflow has occurred

- septic systems, dry wells, or injection wells that received radioactively contaminated liquid effluent

- storage tanks, waste tanks, and/or piping (above or below ground) that held or transported radioactively contaminated fluids and are known to have leaked

- liquid or wet radioactive waste buried on site (i.e., burial under previous regulations found in 10 CFR 20.302 “Method for Obtaining Approval for Proposed Disposal Procedures”, or 10 CFR 20.304 “Disposal by Burial in Soil”, (or the current 10 CFR 20.2002, “Method for Obtaining Approval of Proposed Disposal Procedures”))

- an accident or spill on site where radioactive material was released exterior to a building

- wet bulk waste (e.g., sludge or tailings) stored exterior to buildings or used as backfill

- containerized-liquid waste, stored exterior to buildings, that has leaked

- Medium Potential: if a site has a history of or currently has the following:

  - surface water or atmospheric discharge of radioactive effluents including authorized releases and spills (e.g., releases in compliance with 10 CFR Part 20, Appendix B effluent concentrations or spills)

  - radioactive contamination detected on the roof of a building

  - radioactive contamination detected in the floor cracks or sump of a building

  - an accident or spill on site, where liquid radioactive material was released to the interior of a building

  - the presence of aging unmonitored underground storage tanks or underground piping that held radioactively contaminated fluids, and are not known to have leaked (e.g., unmonitored tanks or piping which could have released radioactivity due to corrosive environmental or service conditions, use of material types potentially susceptible to corrosion [carbon steel], or design flaws)

  - a history of incineration of radioactive waste exterior to onsite buildings

  - dry bulk waste (e.g., sludge or tailings) stored exterior to buildings or used as backfill

  - solid containerized waste, stored exterior to buildings, that has leaked

- Low Potential: if a site has a history of or currently has the following:

- underground storage tanks or underground piping early in their service life that held radioactively contaminated fluids and are not known to have leaked
- dry bulk waste stored inside the buildings
- a sealed-source-only license

The potential for groundwater contamination at any of these sites is conditioned by certain site characteristics, such as depth of groundwater, amount of yearly precipitation, and hydraulic conductivity, and by certain source characteristics such as half-life, solubility, and distribution coefficient.

## F.5 Groundwater Characterization

If existing groundwater contamination is known to be present, or if it is determined that residual radioactivity has a medium to high likelihood of reaching groundwater, radiological surveys of groundwater are expected.

Characterization of groundwater contamination, including all significant radiological constituents, along with inorganic and organic constituents and related parameters, should be adequate to determine the following:

- extent and concentration distribution of contaminants
- source (known or postulated) of radioactive contaminants to groundwater
- background groundwater quality
- rate(s) and direction(s) of contaminated groundwater migration
- location of groundwater plume and concentration profiles (i.e., maximum concentration in the vertical and lateral extent)
- assessment of present and potential future effects of groundwater withdrawal on the migration of groundwater contaminants
- potential safety and environmental issues associated with remediating the surface water and groundwater
- chemical form/speciation of the radionuclides; effect of the non-radiological constituents on the mobility of the radionuclides
- whether the remediation activities and radiation control measures proposed by the licensee are appropriate for the type and amount of radioactive material present in the surface water and groundwater
- whether the licensee's waste management practices are appropriate
- whether the licensee's cost estimates are plausible

Besides licensee process discharges, other mechanisms may affect groundwater. For example, sumps that capture infiltrating groundwater may affect the local groundwater elevation during pumping. In some situations, sumps collect groundwater at the lowest elevation of a building, with pumping going on continuously. Such pumping has been shown to affect the local groundwater elevation (i.e., cone of depression).

Typical analytical parameters include gross alpha particle activity, gross beta particle activity, specific radionuclide concentrations, gamma spectrum analysis for all gamma-emitting radionuclides suspected to be present, sulfate, chloride, carbonate, alkalinity, nitrate, total dissolved solids, total organic carbon, Eh, pH, calcium, sodium, potassium, iron, and dissolved oxygen. Additional analytical parameters may be necessary to characterize any suspected contamination. Other regulatory agencies that have jurisdiction over the decommissioning effort may require characterization of the non-radiological constituents and related parameters including or in addition to those listed above. Therefore, licensees should contact Federal, State, or local government bodies responsible for regulating groundwater.

The licensee should determine the extent of contamination and background groundwater quality based on groundwater monitoring data from a suitable monitoring well network. The reference section lists the following guidance documents on acceptable groundwater monitoring techniques: Korte and Ealey (1983), Korte and Kearl (1984), NUREG-6948, NUREG/CR-7221, and NUREG/CR-1388, USGS (1977, 1996 and 2018), EPA (1977, 1980, 1985, 1986, and 2009a), and ASTM D5092/D5092M (2016a). The actual number, location, and design of monitoring wells depend on the size of the contaminated area, the type and extent of contaminants, the background groundwater quality, the hydrogeologic system, and the objectives of the monitoring program. For example, if the objective of monitoring is only to indicate the presence of groundwater contamination, the licensee will need relatively few downgradient and upgradient monitoring wells. In contrast, if the objective is to develop a detailed characterization of the distribution of constituents within a complex aquifer as the design basis for a corrective action program, a large number of suitably designed and installed monitoring wells and well points may be necessary.

Planned site characterization activities should be flexible enough to allow for the installation of additional monitoring wells during the characterization effort if either (1) preliminary characterization indicates contamination where previously unanticipated, or (2) there is a need to delineate the vertical or lateral extent of contaminant plumes. Monitoring well locations, contaminant concentrations, and contaminant sources should be plotted on a map (or a series of maps for multiple contaminants) to show the relationship among contamination, sources, hydrogeologic features and boundary conditions, and property boundaries. At sites with significant vertical migration of contaminants, the Decommissioning Plan (DP) should also provide hydrogeologic cross sections that depict the vertical distribution of contaminants in groundwater. The vertical exaggeration of the cross-sections should not exceed 10 times.

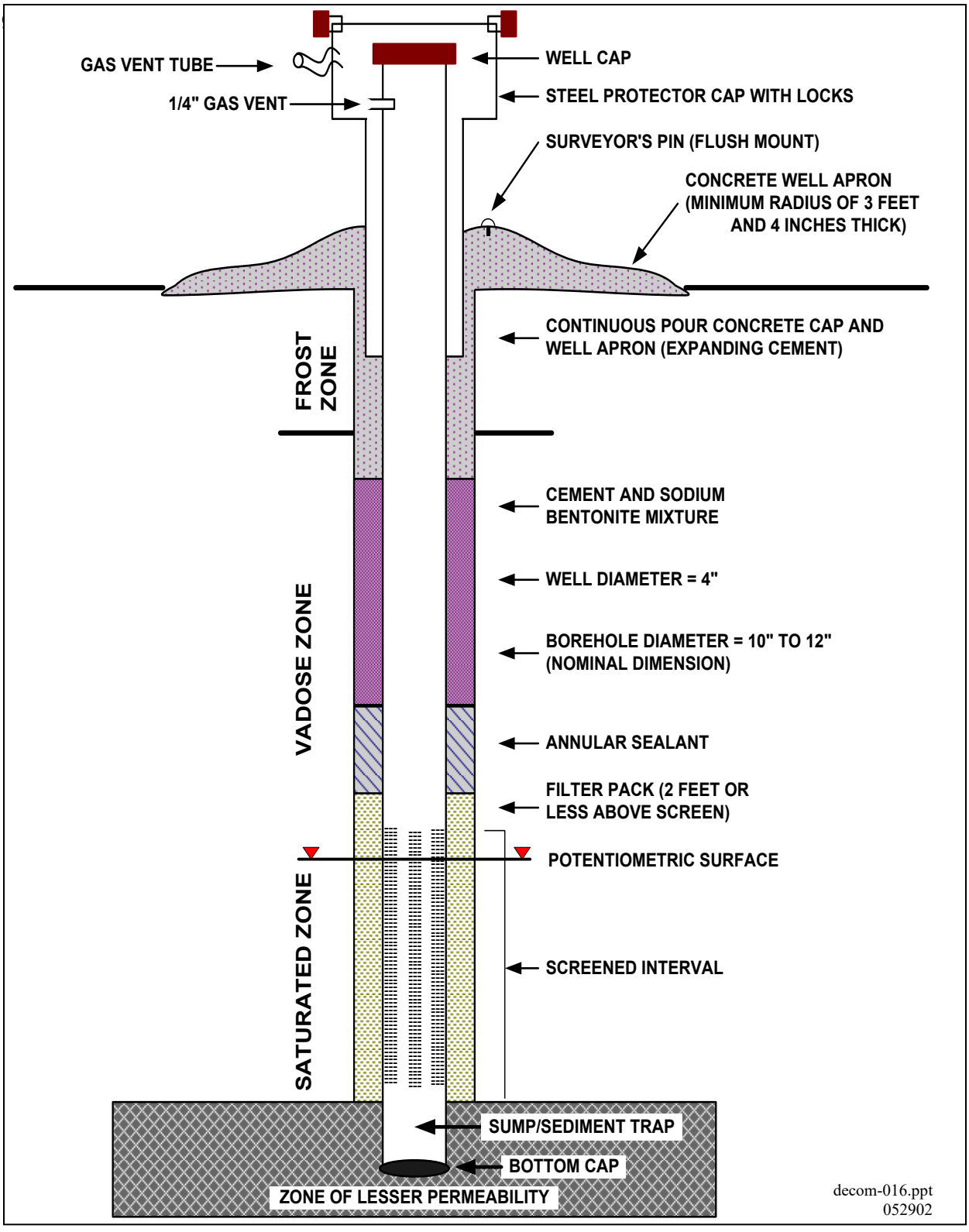


**Figure F.1 General Monitoring Well Cross Section (Adapted from Figure 3 [NRC, 1994])**

The DP should also describe the groundwater characterization program used to characterize the extent and distribution of contaminants in the groundwater. Depending on the complexity of the site, the DP can include the detailed information as described below or can summarize this information and then reference the documents containing the supporting details. The description should provide monitoring well completion diagrams explaining elevation, internal and external dimensions, type of casings, type of backfill and seal, type of the screen and its location and size, borehole diameter and elevation and depth of hole, and type and dimension of riser pipe and other necessary information on the wells. Figure F.1 illustrates an acceptable generic design for well completion. The reference section lists the following documents that may contain useful information on monitoring well installation for specific objectives and geohydrologic conditions: ASTM (2016a, 2016b, and 2018a), EPA (1991a, and 2013), and USGS (1996). ASTM (2018b) may provide useful information on how to remove monitoring or remediation wells from service without creating a conduit for contaminant migration from the surface or between subsurface geologic units.

The DP should document or reference the sampling techniques, methodology, and procedures. Site characterization procedures and methods should generally adhere to national practices and standards (e.g., American Society for Testing and Materials (ASTM), the USGS, EPA, U.S. Department of Energy Environmental Monitoring Laboratory (DOE/EML), and National Institute of Standards and Technology (NIST)). The DP should identify specific analytical methods that conform to generally accepted protocols and methods, such as those endorsed by NIST and DOE/EML, or other methods established through a comprehensive peer review and recommendation process (e.g., American National Standards Institute (ANSI)/American Society of Mechanical Engineers (ASME), "Quality Assurance Program Requirements for Nuclear Facilities," 1986). Korte and Kearl, "Procedures for the Collection and Preservation of Groundwater and Surface Water Samples and for the Installation of Monitoring Wells," 1984, provides forms for documenting well summary information, samples, chain of custody, QA information for field chemical analyses, and sample location and identifier.

The site characterization program should include sufficient sampling and analysis of groundwater samples collected upgradient from the site to develop a representative characterization of background groundwater quality. Background groundwater quality should not exhibit any influence from contaminants released by the site and should be representative of the quality of groundwater that would exist if the site had not been contaminated. The site characterization should also assess any temporal or spatial variations in background groundwater quality. If sources of contamination other than the site are present, the licensee should evaluate the potential impact of such sources to determine the degree of their groundwater contamination.

## F.6 Monitoring Practices and Procedures

Depending on the complexity of the site, the DP can include the detailed information as described below, or the DP can summarize this information and then reference the documents containing the supporting details.

The site characterization should include a description of all surface water and groundwater characterization activities, methods, and monitoring installations sufficient to demonstrate that the methods and devices provided data that are representative of site conditions. It should also describe the monitoring practices, procedures, and QA programs used to collect water quality data. Monitoring well descriptions, for example, should include location, elevation, screened interval(s), depth, construction and completion details, and the hydrologic units monitored.

Aquifer test descriptions should include testing configuration, test results, and a discussion of the assumptions, analytical techniques, test procedures, pretesting baseline conditions, limitations, errors in measurements, and final results. The description of the water quality sampling and analysis program should include or reference the procedures for sampling, preserving, storing, and analyzing the samples, including QA/QC protocols implemented. All methods should be consistent with current standard methods and practices (e.g., ASTM, USGS, EPA, NIST, and ANSI/ASME). The reference section lists the following documents containing additional guidance on applicable methods for sampling and analyzing water quality samples: Korte and Ealey (1983); Korte and Kearl (1984); DOE (1988 and 1993); ANSI/ASME (1986); EPA (1977, 1985, 1986, 1987a, 1991b); and NUREG-1293, NUREG-1383, and Regulatory Guide 4.15 (NRC, 1979). The licensee should document and explain any deviations from the standard methods used.

## F.7 Monitoring Network and Sampling Frequencies

The monitoring wells are usually placed in the following critical locations (NRC, 2007):

(1) Source areas,

(2) Within and immediately down-gradient of the source area;

(3) Fringe portions and boundary of the plume;

(4) Regulatory compliance boundaries.

To address the effect of physical/chemical heterogeneity on contaminant fate and transport, certain monitoring wells should be placed in relatively high transmissive zones with highest contaminant concentrations in the targeted monitoring areas; monitoring wells may need to be placed in locations that will reduce uncertainties with respect to geochemical conditions, and aquifer hydraulic properties. NUREG/CR-6948, NUREG/CR-7221, EPA (2009a), and Barcelona et al. (1989) provide information on how to design and implement a more optimal groundwater monitoring well network to refine site conceptual and numerical models and improve the capabilities for detection of contamination.

It is important to note that operational environmental monitoring of groundwater, although adequate for its intended purpose, may not be adequate for site characterization and to support dose assessments. To support site characterization and dose assessments, information supplied by licensees may need to address the types and movement of radioactive contamination in groundwater at the facility, as well as the magnitude and extent of this contamination. The actual number, location, and design of monitoring wells depend on the size of the contaminated area, the nature and extent of contamination, the background quality, hydrogeologic system, and the objectives of the monitoring program. For example, if the only objective of monitoring is to indicate the presence of groundwater contamination, relatively few downgradient and upgradient monitoring wells are needed. In contrast, if the objective is to develop a detailed characterization of the distribution of constituents within a complex aquifer as the design basis for a corrective action program, a relatively large number of suitably designed and installed monitoring wells may be necessary. Power reactors normally have groundwater monitoring programs as part of their radiological environmental monitoring programs (REMPs). Although data derived from a REMP may provide useful information, the data still tend to be insufficient to allow the staff to fully understand the fates and transport of radioactive materials

in the subsurface environment at the facility, as well as the magnitude and extent of this contamination (see for example, RG 1.185, Post-Shutdown Decommissioning Activities Report, lessons learned on groundwater monitoring). Therefore, a licensee may need to gather additional data to address this lack of understanding.

If remediation is necessary, remedial performance confirmation monitoring may be conducted to demonstrate that remediation is occurring according to expectations. Although remedial expectations and, consequently, appropriate performance monitoring analyses are site-specific in nature, reduction in contaminant concentrations to specified levels is generally expected as selected remedies are being performed (Pope et al., 2004). Data analysis is useful in assessing progress toward contaminant reduction objectives include evaluation of temporal trends in contaminant concentrations or mass, comparisons of observed contaminant distributions with predictions or predefined targets, among other approaches. Evaluations of adequate progress toward restoration objectives are sometimes difficult due, in large measure, to subsurface spatial variability and to a lesser extent, measurement variability. This will often necessitate optimization of the employed remedial remedy, with relatively dense monitoring networks to reduce uncertainty to acceptable levels.

The remedial action objective of attaining permitted standards, such as groundwater protection standards, or DCGLs should be demonstrated before monitoring and the license are terminated to ensure that the required standards are actually achieved in the long-term (Pope et al., 2004). The demonstration of the attainment of cleanup objectives should include sufficient verification monitoring once the standards are achieved to evaluate the effects of variations caused by active remedy. The length of the verification period should be based on site-specific conditions and on objective statistical analyses of the data. Site-specific conditions to consider include the response times of the hydrogeological system to seasonal or annual variations. Statistical methods useful in these evaluations include analyses of temporal trends in contaminant concentrations and comparisons with the specified concentration standards (e.g., Cohen et al., 1994; EPA, 1992; and ITRC, 2013).

The licensee should establish surface water and groundwater quality and water levels on a set frequency, based on site-specific considerations. For sites with extensive groundwater contamination, the licensee should design and install a network of monitoring wells to provide a high probability of detecting and characterizing existing contamination and determining background groundwater quality. It should measure groundwater levels in piezometers and monitoring wells that provide a sufficiently accurate change in hydraulic head to determine the horizontal hydraulic gradient within the uppermost aquifer and vertical hydraulic gradient with adjacent units. It should measure water levels at least on a quarterly basis for a minimum of one (1) year to determine temporal variations in the hydraulic conditions. After this period, it should adjust the frequency of water level measurements to reflect the anticipated impact on hydraulic heads by site-specific events and conditions (e.g., tides, rises in river stage and bank storage, increased precipitation, water year variations). The initial monitoring program should be designed so that it can be integrated into the programs for operational and post-operational periods. The reference section lists the following documents containing applicable methods for groundwater sampling and for measuring water levels: EPA (1977, 1985, 1986, and 1987a), USGS (1977, 2018), and Korte and Kearl (1984), and NUREG-1388.

The licensee should determine the sampling frequency for evaluating spatial and temporal variations in groundwater quality, including radiological and associated non-radiological constituents, based on the site-specific temporal variations in flow directions and hydrogeochemical conditions. After an initial sampling period in which each monitoring well is

sampled at a frequency to establish site-dependent temporal variations throughout a year, it should collect and analyze representative samples generally on a quarterly basis from key monitoring wells to obtain representative estimates of the temporal variation of water quality in the uppermost aquifer and adjacent units. After this initial period, it should adjust the sampling frequency to reflect variations in the hydraulic gradient and hydrochemistry. If concentrations of principal radiological constituents change by more than about 10–20 percent between sampling events, it should increase the frequency of sampling in an attempt to characterize the temporal variability of groundwater quality. For unconfined groundwater systems, less than biannual sampling schedules may not capture important temporal variations. For most sites, it should be sufficient to sample on a quarterly basis (i.e., one sample per well per calendar quarter) to characterize temporal changes in water quality. More frequent sampling, such as bimonthly may be necessary, however, especially at sites involving offsite or potential offsite contamination of groundwater resources with more reactive, geochemical constituents in a dynamic hydrological system (Barcelona et al., 1989). The sampling frequency needs to be evaluated with respect to the monitoring objective and time-frame over which the monitoring network will be conducted. NUREG/CR-6948, NUREG/CR-7221, EPA (2009a), and Barcelona et al. (1989) include acceptable approaches for assessing frequency of groundwater sampling.

Quarterly sampling of surface water and sediments should be sufficient at most sites. This sampling should be supplemented by additional sampling to characterize the surface water system at representative low- or high-stage flow conditions (e.g., minimum annual, 7-day average low flow or maximum annual, 7-day average high flow). The licensee should use this information to bound the existing and projected impacts of the release of contamination on adjacent surface water bodies.

Note that the incremental benefit of sampling decreases with increasing sampling frequency when there is an autocorrelation in the data. If this autocorrelation is large, a relative low sampling frequency is necessary to avoid sampling redundancy, and the total length of the sampling period must increase to achieve a sufficient return on sampling. The adequacy of the sampling frequency needs to be interpreted or viewed in terms of the time horizon of the sampling program and monitoring objectives (Barcelona et al., 1989).

## F.8 Surface Water and Sediments

Surface water can include ponds, creeks, streams, rivers, lakes, coastal tidal waters, oceans, and other bodies of water. Note that certain ditches and intermittently flowing streams qualify as surface water. The licensee should evaluate the need for surface water samples on a case-by-case basis. It should base its surveys for water on appropriate environmental standards for water sampling. If the body of water is included in a larger survey unit, then it should take sediment samples at sample locations selected by the normal method, without taking the body of water into consideration. In addition to the discussion below, MARSSIM (Sections 3.6.3.3 and 5.3.3.3) provides some guidance on evaluating the likelihood for the release of radionuclides into surface water and sediments and on concerns related to characterization and sampling.

Onsite and offsite normal effluent discharges to surface water may be permitted under the license or by wastewater and stormwater permits and registrations. Radioactive Effluent Reports track these releases and provide dose assessment results for the appropriate scenario pathway. In addition, Radiological Environmental Monitoring Reports record radiological results of offsite sampling of water and sediments at discharge locations. For effluent discharges to onsite lakes or ponds at power plants, leakage to the groundwater may occur, though it need

not be reported again during operations (NRC, 2017b). However, at decommissioning, the residual radioactive in the lake or pond sediments and any leakage to the groundwater must be treated as residual radioactivity that needs to be dispositioned in accordance with the guidance in this NUREG. For offsite effluent discharges, offsite decommissioning activities are not required by NRC regulations. State and local entities may require different treatment of offsite areas that are contaminated by normal effluent discharges. For NRC, however, characterization may be needed to assess environmental impacts as part of the environmental assessment or impact statement. Similarly, groundwater seepage of onsite residual contamination to offsite surface waters must be incorporated into environment assessments or impact statements. Any onsite groundwater contamination during decommissioning must be incorporated into the dose assessment and site release criteria.

For sites that are located near surface water streams and could reasonably affect surface water pathways, the site characterization program should establish background surface water quality by sampling upstream of the site being studied or areas unaffected by any known activity at the site. Water should be collected as grab samples from the stream bank in a well-mixed zone. Depending on the significance and the potential for surface water contamination, it may be necessary for certain sites to collect stratified samples from the surface water to determine the distribution of contaminants within the water column. Surface water quality sampling should be accompanied by at least one round of stream sediment quality sampling to assess the relationship between the composition of the dissolved solids, the suspended sediment, and the bedload sediment fractions. The licensee should determine water levels and discharge rates of the stream at the time samples are collected and should also consider the effects of variability of the surface water flow rate. Based on the results of the HSA and preliminary investigation surveys, it should conduct surface scans for gamma activity in areas likely to contain residual activity (e.g., along the banks). Korte and Kearl (1984) and USGS (1977, 2018) describe applicable methods for surface water and sediment sampling. In addition, Fleischhauer and Engelder (1984) present suggested procedures for stream sediment sampling. The EPA guidance documents mentioned above are also applicable. In some cases, the Radiological Environmental Monitoring Program (REMP) data from a facility's operating period may provide useful information to support the characterization program, although there are limitations of the use of REMP data to support site characterization.

The licensee should conduct surface water sampling in areas of runoff from active operations. In case of direct discharge into a stream, it should monitor and sample the outfall and the stream upstream and downstream from the outfall. It should conduct preliminary characterization of the contamination levels by measuring gross alpha and total beta particle activity (total and dissolved) and by obtaining a gamma spectrum for surface water samples. It should be noted that determination of gross alpha activity (and low-energy beta emitters, as well) may be of limited value for samples containing elevated total or dissolved solids concentrations because of sample attenuation. In such instances, gamma spectroscopy might be the only recourse. Specific radionuclide analysis may be needed, depending on the level of activities and type of radionuclides. Non-radiological parameters, such as specific conductance, pH, and total organic carbon, may be used as surrogate indicators of potential contamination, provided a clear relationship is established between radionuclide concentration and the level of the surrogate. Additional analysis for other parameters like volatile and semivolatile compounds, chelating agents, pesticides, and polychlorinated biphenyls may also be necessary if they affect the mobility of radiological constituents and to evaluate the potential environmental effects of decommissioning.

The licensee should carefully record each of the surface water and sediment sampling locations on the appropriate survey form. It can also use surface water flow models to assist in estimating contaminant concentrations or migration rates.

## F.9 Geochemical Conditions

The licensee should also describe geochemical conditions at the site and their association with groundwater and contaminants, giving consideration specifically to geochemical conditions that enhance or retard contaminant transport. Geochemical data should include information on solid composition, buffering capacity, redox potential, pH, sorption (represented as a range of distribution coefficients ($K_d$) for each radiological constituent), and other relevant geochemical data. Piper and Stiff diagrams may be useful for visualizing the geochemistry of the water. In some cases, it may be necessary for licensees (or responsible parties) to use appropriate geochemical codes to understand and quantify geochemical mechanisms that significantly affect transport of radiological and non-radiological contaminants and their potential fate (e.g., geochemical speciation codes such as PHREEQC (Parkhurst and Appelo, 2013) are typically used for this purpose)).

In general, licensees or responsible parties should estimate the values of $K_d$ through laboratory column or batch sorption measurements considering site-specific conditions, if found to be important to estimating dose. If not found to be important to dose, site-specific $K_d$ measurements may not be necessary and generic ranges may be used from published tables (e.g., Sheppard and Thibault, 1990; EPA, 1999, 2004, 2005; and Yu et al., 2015). Licensees (or responsible parties) may use appropriate geochemical codes to understand and quantify geochemical mechanisms that significantly affect transport of radiological and non-radiological contaminants and their potential fate (e.g., MINTEQ (EPA 1984); EQ3/6 (Daveler and Woolery, 1992); PHREEQC (Parkhurst and Appelo, 2013)). For complex sites with the possibility of transient variable geochemical conditions, the use of surface complexation models in geochemical speciation models may be needed to adequately represent the variability of $K_d$ (e.g., NRC, 2003a).

Appendix I of this volume discusses additional information on groundwater parameters necessary for dose modeling.

## F.10 Surface Water and Groundwater Models to Support Dose Modeling

Appendix I contains additional information on hydrogeological conceptual models found in commonly used decommissioning dose modeling codes such as DandD[3] (NRC, 2001) and RESRAD-ONSITE (Kamboj et al., 2018). The inherent assumptions and limitations of these models is addressed. RESRAD-ONSITE does not consider existing groundwater contamination and only addresses the potential, future transport of residual radioactivity and contamination of ground and surface water and associated doses.[4]

In cases where existing groundwater contamination is present, it is likely necessary to construct a more complicated groundwater model to support development of DCGLs. Additionally, in

3 Decommissioning codes such as DandD can be obtained from NRC's RAMP (Radiation Protection Computer Code Analysis and Maintenance Program).

4 RESRAD-OFFSITE 3.2 considers sources located below the water table, although guidance on potential use of RESRAD-OFFSITE to consider existing groundwater contamination has not yet been developed. A future revision to this volume may include evaluation of this tool and its efficacy in considering existing residual radioactivity in groundwater.

these cases some allowance for the existing groundwater contamination must be made (e.g., fractionation of the release criteria to account for multiple contaminated media) to ensure the potential dose is not underestimated. Although a licensee always has the option of assessing the peak dose within the 1000-year compliance period based on more realistic modeling, in many cases it may be more straightforward to perform more bounding type analyses to simplify the decommissioning process if compliance can be easily demonstrated.

As a joint effort, the NRC, EPA, and DOE have developed specific guidance on selecting and applying surface water and groundwater models (EPA 1994a, b, c). The reference section lists other documents containing supporting details (NUREG/CR-6805 and NUREG/CR-6948; ASTM D5447-17 (2018c), NCRP 1985, 1996; EPA 1987a, 1987b, 2009b; NAS 1999).

As previously discussed, model simplification is the process for reducing the complexity of a numerical model into a simpler numerical model while still maintaining the validity of the simulation results. Although specific computer codes may be discussed or referenced in this guidance, the NRC does not endorse the use of any particular code or modeling software package for analyzing the performance of a decommissioning or disposal site. However, it would be useful to discuss commonly used codes for developing groundwater models by NRC licensees and to know which codes are better suited for constructing complex models and which codes usually provide good results when used to construct simple models. A discussion of commonly used codes and a comparison of the model results obtained by codes used to construct complex models and the results from codes used for simple models should provide for a better understanding of the model simplification process and information on how model simplifications can affect the modeling results.

### F.10.1 Codes for Development of More Complex Models

The MODFLOW and MT3DMS codes are commonly used to simulate multi-dimensional flow and transport in the subsurface. These codes can simulate three-dimensional flow and transport, and therefore can generally better project contaminant concentrations at various points of exposure compared to codes that are only able to simulate flow and transport in one-dimension. Of course, sufficient data must be available to construct these more complex models, or the additional code capabilities are of limited value. In some cases, it may be preferable or necessary to use simpler codes to simulate groundwater flow and transport when adequate safety margin is available and more conservative assumptions can be made, or when data or resources are limited making it difficult for an analyst to construct a more complex model of the site. However, when additional complexity is warranted there are several off the shelf codes available to assist with constructing these more complex three-dimensional groundwater flow and transport models.

The MODFLOW code is a groundwater flow modeling code developed by the USGS that solves the groundwater flow equation in up to three dimensions using finite-difference approximations. The code is public domain free software written primarily in FORTRAN and can be compiled and run on Microsoft Windows or Unix-like operating systems. The first version of MODFLOW was published in 1984. It has a modular structure and can be modified to address different applications or problems. MODFLOW-2005 has many new capabilities compared to older versions (Harbaugh, 2005). MODFLOW 6 is the current core MODFLOW version distributed by the USGS (Langevin, et al., 2017; Hughes, et al., 2017). The previous core version, MODFLOW-2005, is actively maintained, supported, and able to handle supplemental MODFLOW features not yet updated for MODFLOW 6.

The MODFLOW-2005 code simulates steady and non-steady flow in an irregularly shaped flow system in which aquifer layers can be confined, unconfined, or a combination of confined and unconfined. Flow from external stresses, such as flow to wells, areal recharge, evapotranspiration, flow to drains, and flow through river beds, can be simulated. Hydraulic conductivities or transmissivities for any layer may differ spatially and be anisotropic (restricted to having the principal directions aligned with the grid axes), and the storage coefficient may be heterogeneous. Specified head and specified flux boundaries can be simulated as can a head dependent flux across the model's outer boundary. MODFLOW 6 uses a new format of blocks and keywords for input of model data and is written using an object-oriented design. MODFLOW 6 uses a control-volume finite difference approach that can utilize structured or unstructured grids; MODLFOW-2005 requires a Cartesian grid system. MODFLOW 6 presently supports one type of process model — the Ground Water Flow Model (GWF). Multiple GWF models can be coupled and concurrently simulated, such as nesting of models at different scales. Other models may be added in the future that can seamlessly couple with the GWF, such as a groundwater transport model, a surface-water model, and a pipe network model. The new MODFLOW 6 framework will enable the ability to solve multiple, tightly coupled, numerical models in a single system of equations, or simplify the computational effort by loosely coupling the different models. For continuity, licensees may choose to continue using MODFLOW-2005; but if starting fresh or the site has hydrogeologically-significant features that exhibit as complex geometries, licensees should consider using MODFLOW 6.

Other options exist for sites with complex geometries. Codes developed using finite volume or finite element representations may be needed to better represent the geometries of the system. A finite volume code in the public domain that can handle a wide range of site characteristics and geometries is TOUGH2 (Pruess, et al., 2012). The PORFLOW™ [5] finite-element code has been used by DOE due to its ability to consider radioactive decay, can solve problems involving transient or steady state fluid flow, and mass transport in multi-phase, variably saturated, porous or fractured media. The porous/fractured media may be anisotropic and heterogeneous, arbitrary sources (injection or pumping wells) may be present and, chemical reactions or radioactive decay may take place. The geometry may be 2D or 3D, Cartesian or Cylindrical and the mesh may be structured or unstructured.

The MT3DMS code is a modular three-dimensional transport model for the simulation of advection, dispersion, and chemical reactions of dissolved constituents in groundwater systems (Bedekar, et al., 2016). The MT3DMS code uses a modular structure similar to the structure utilized by MODFLOW. The MT3DMS code is used in conjunction with MODFLOW in a two-step flow and transport simulation. Heads and cell-by-cell flux terms are computed by MODFLOW during the flow simulation and are written to a specially formatted file. This file is then read by MT3DMS and utilized as the flow field for the transport portion of the simulation. The MT3DMS code differs from MT3D in that it allows for multi-species transport, supports additional solvers, and allows for cell-by-cell input of all model parameters.

### F.10.2 Codes Typically Used in Decommissioning

The RESRAD-ONSITE (Kamboj et al., 2018) and RESRAD-OFFSITE (NRC, 2020) codes are commonly used for calculating groundwater concentrations and groundwater-dependent dose for decommissioning dose assessments. Details of the RESRAD-ONSITE groundwater models are discussed in Appendix I.

5 PORFLOW was developed by Analytic & Computational Research, Inc. (ACRi, 2008).

The RESRAD-ONSITE code includes a mass balance and non-dispersion model to compute groundwater concentrations at an onsite well. Neither the RESRAD-ONSITE mass balance, nor the non-dispersion model, considers dispersion, while RESRAD-OFFSITE can consider advection *and dispersion* in calculating groundwater concentrations at a receptor well. The groundwater transport model in RESRAD-OFFSITE considers 1-D advection (straight or curved flow path), along with 3-D dispersive transport in the saturated zone. Likewise, while only 1-D advection is considered in RESRAD-ONSITE, RESRAD-OFFSITE considers 1-D advection, and 1-D dispersive transport in the unsaturated zone. Furthermore, the unsaturated zone, saturated zone, and contaminated zone[6] can be subdivided into smaller zones to increase the accuracy of transport simulations.[7]

Both RESRAD-ONSITE and RESRAD-OFFSITE have the capability to consider variable transport rates of progeny created during transport in groundwater. The RESRAD-OFFSITE code has two groundwater transport algorithms: the first algorithm considers variable transport rates of parents and progeny, and the second algorithm models longitudinal dispersion[8]. When either variable transport rates or dispersion is clearly dominant, the RESRAD-OFFSITE user should choose the transport algorithm that is most important to increase computational efficiency. When both the longitudinal dispersion and the variable transport rates are important, the user has the option of subdividing the transport pathway into a number of subzones to more accurately simulate the transport of progeny in transport, although this approach may significantly increase computation times. Only the zone where the progeny atoms are created would not consider both processes (longitudinal dispersion and variable transport rates of parents and daughters).

*Benchmarking of RESRAD-ONSITE with RESRAD-OFFSITE (onsite simulation)*

Code developers benchmarked RESRAD-OFFSITE against RESRAD-ONSITE (Yu, 2006) using an earlier version of the RESRAD-OFFSITE code (i.e., benchmarking was conducted prior to the 2007 release of RESRAD-OFFSITE 2.0). The results of the benchmarking exercises showed that RESRAD-OFFSITE could mimic the results of RESRAD-ONSITE when certain parameters were changed consistent with the RESRAD-ONSITE conceptual model. Notable differences between initial simulations run with RESRAD-ONSITE and RESRAD-OFFSITE included travel times to the point of compliance that were attributable to differences in use of porosity in the transport calculations (i.e., effective porosity is used in RESRAD-ONSITE, while total porosity is used in RESRAD-OFFSITE). Another noteworthy difference in results was observed for the water-dependent pathways due to accumulation of radioactivity in soil from application of contaminated irrigation water that is considered in RESRAD-OFFSITE but is not considered in RESRAD-ONSITE.

6 The capability to more accurately simulate transport through the contaminated zone through use of sub-zones was added in RESRAD-OFFSITE, Version 3.0

7 It is important to note that RESRAD-OFFSITE also has the capability of mimicking the RESRAD code for calculation of doses to an onsite receptor. However, reference to RESRAD-OFFSITE models and calculations in this section pertain to just the offsite capabilities, and not the onsite dose calculations.

8 Both RESRAD-OFFSITE transport algorithms account for transverse dispersion.

*Benchmarking of RESRAD-OFFSITE with Other Codes*

The RESRAD-OFFSITE[9] code was also benchmarked with several other codes as detailed in Gnanapragasam et al. (2000). The benchmarking study is instructive as it provides extensive discussion on the differences in modeling results for more complex and less complex models. Specifically, the model comparison study illustrates how differences between codes in their treatment of longitudinal and transverse dispersion and differences in ability to model variable transport rates of parents and daughters impact the results. The radionuclides considered in the benchmarking study included relatively short-lived Sr-90, and U-234 decay chain members (U-234, Th-230, Ra-226, Pb-210, and Po-210). Source loading spanned 200 years for Sr-90 and 100 years for U-234 (at a constant rate of 0.88 Ci/y). The RESRAD-OFFSITE, PRESTO, MMSOILS, and MEPAS codes were the four primary codes tested in this study. They have different capabilities with respect to simulation of dispersion and variable transport rates of progeny. Increasing dispersivity with distance, and variable pore water velocity along the flow path were also considered in this study.

The results of the comparison showed markedly different results for PRESTO at increasing distance from the source, and at longer times after source loading ceases (when longitudinal dispersion is important). The PRESTO longitudinal profiles and breakthrough curve results are different from other codes because PRESTO is the only code of the four codes tested that does not consider longitudinal dispersion. However, at shorter distances and for constant sources, the differences between PRESTO and the other codes are less significant.

Another important difference between the codes tested is related to the calculation of retardation. Because RESRAD-OFFSITE is the only code to consider immobile pore water in the calculation of the retardation factor, the calculated retardation factors can be significantly higher for radionuclides with low retardation factors and if the effective and total porosity are significantly different (e.g., in the Gnanapragasam et al. (2000) study RESRAD-OFFSITE calculated retardation factors are higher by a value of 0.56 (unitless) for all radionuclides due to the difference in total and effective porosity at 0.39 and 0.25, respectively). Differences in retardation factors also have a more significant impact on relatively short-lived[10] radionuclides.

All of the codes are slightly different when it comes to important processes for progeny created in transport (i.e., longitudinal dispersion and consideration of variable transport rates of parents and progeny). The PRESTO code considers neither longitudinal dispersion, nor variable transport rates of progeny. The version of RESRAD-OFFSITE tested in the study only considered variable transport rates of daughters but not longitudinal dispersion of progeny (additional features were added to RESRAD-OFFSITE in subsequent versions as described above). When longitudinal dispersion is not considered, a sharp front is observed in the breakthrough curves of progeny; however, if variable transport rates of progeny are considered, as is the case for RESRAD-OFFSITE, higher retardation factors of progeny (e.g., Th-230 has a retardation factor of 198 compared to a retardation factor of 48 for U-234) can lead to long tails in the breakthrough curves. RESRAD-OFFSITE's capability of modeling variable transport rates of progeny also leads to lower peak concentrations of radionuclides such as Th-230, which has a higher retardation factor than its parent. On the other hand, higher peak concentrations of radionuclides such as Ra-226, Pb-210, and Po-210 result due to (1) the high retardation factor of Th-230, and (2) the lower retardation factors of the radionuclides compared to their

9 The Gnanapragasam et al. (2000) reference refers to what is now called RESRAD-OFFSITE as just RESRAD. Reference to RESRAD-OFFSITE in this section is to an earlier prototype version of RESRAD-OFFSITE.

10 Short-lived with respect to the observation time or transport time at the point of observation.

immediate parents. Improvements to the accuracy of the breakthrough curves were observed when the aquifer was broken up into smaller segments and longitudinal dispersion and variable transport rates of progeny (compared to parents) was considered in all segments with the exception of the segment of transformation (where either longitudinal dispersion or variable transport rates of the progeny were considered). Presumably, this study led to improvements in the progeny transport models for some of the evaluated codes, including more recent versions of RESRAD-OFFSITE.

The various codes studied also differ with respect to the consideration of dilution in a pumping well (i.e., some codes consider just aquifer concentrations, while other codes such as RESRAD-OFFSITE and PRESTO can consider dilution in a well). Depending on the location of the well and other factors, the impact of pumping may be more or less significant. For example, the study concludes that at exposure points far away from the source, transverse dispersion can be expected to distribute contamination nearly uniformly near the well given the size of the plume in relation to the capture zone. On the other hand, if the well is located close to the source and the plume is narrow or not well-developed, then pumping may lead to dilution of the narrow contaminant plume via mixing with clean water in the larger (than plume) volume of water pulled in by the well.

A prototype version of RESRAD-OFFSITE was also evaluated in a study conducted by the Biospheric Model Validation Study II Working Group on Uranium Mill Tailings (BIOMOVS II 1996). The predictions of well water concentrations and offsite soil accumulation made by the prototype version of RESRAD-OFFSITE were stated by code developers to compare well with the predictions of other codes participating in the study (Yu, 2006).

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# APPENDIX G

# SPECIAL ISSUES ASSOCIATED WITH DOSE MODELING, CHARACTERIZATION, AND SURVEY

# G.1 Introduction

There are several special situations during the decommissioning process that are not, or are only minimally, addressed in the regulatory guidance of the NRC and in NUREG-1575, "Multi-Agency Radiological Survey and Site Investigation Manual (MARSSIM), Revision 1, issued August 2000. In these situations, licensees may need to perform characterization and a FSS to demonstrate compliance with the license termination criteria in 10 CFR Part 20, "Standards for Protection Against Radiation," Subpart E, "Radiological Criteria for License Termination." As part of its review and approval of DPs and LTPs, the NRC staff at this time evaluates these special situations on a case-by-case basis. The NRC may develop additional guidance in the future that covers these special situations and will include them in revisions to the consolidated guidance.

This Appendix G applies, either in total or in part, to Decommissioning Groups 4–7.

# G.2 Surveys for Special Situations in Buildings

The survey method described thus far in this volume (e.g., Chapter 4 and Appendix A) applies to simple ideal geometries in a straightforward manner; however, there are likely to be some additional special situations at actual sites that will need further consideration. For each situation discussed below, it is assumed that the HSA and minimal site characterization have located and given a rough estimate of the concentration of residual radioactivity present.

## G.2.1 Structures Versus Equipment

### *G.2.1.1 Background*

The NRC staff acknowledges that the relationship between the LTR for unrestricted use of a site (dose criteria of 0.25 mSv/y (25 mrem/y) and ALARA found in 10 CFR 20.1402), and existing guidance for unrestricted releases of solid materials from a site on a case-by-case basis under 10 CFR 20.2002 may have been unclear. In particular, the criteria for the LTR and for releases of solid materials with small or no amounts of residual contamination before license termination are different. Consistent with the LTR, once a site meets the radiological criteria for unrestricted use and the NRC terminates the license, solid material may be removed from a site. However, before license termination, material cannot be removed from the site for unrestricted use unless it meets either (1) criteria already approved for the licensed facility (e.g., in a license condition), for surficially contaminated materials, or (2) the few mrem/y criterion for the case-by-case approach for volumetrically contaminated materials (see Section 15.11 in Volume 1 of this NUREG). One rationale for the difference in criteria is that the technical basis for the LTR assumes that individuals are generally exposed to residual radioactivity at a single location (the site), while, for releases of solid material, an individual may be exposed to materials through several exposure scenarios at offsite locations. For more information about the relationship between the LTR and the case-by-case approaches to release of solid materials from a site, see the LTR Analysis Commission Paper, SECY-03-0069, "Results of the License Termination Rule Analysis," dated May 2, 2003, and the associated Regulatory Issue Summary 2004-08, "Results of the License Termination Rule Analysis," issued May 2004.

This section focuses on compliance with the LTR, in particular, the building structure-related materials that may be left on site at license termination and the criteria that should apply. Section 15.11 of Volume 1 of this NUREG provides more information about current approaches to releases of solid material before license termination.

### *G.2.1.2 Implementation*

The LTR applies to building structures that remain in place after decommissioning and does not apply to releases of equipment from the facility before license termination. If licensees elect to dismantle building structures and dispose of the associated materials off site (in accordance with applicable regulatory requirements), rather than leave the building structures in place (for unrestricted use), the LTR does not apply to the associated materials moved off site before license termination. Materials licensees may release equipment and building structure deconstruction and dismantlement materials in accordance with existing license conditions. Reactor licensees (licensed under 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities") may release equipment and building structure deconstruction and dismantlement materials in accordance with the guidance in Inspection and Enforcement Circular 81-07, "Control of Radioactively Contaminated Material," dated May 14, 1981; Information Notice 85-92, "Surveys of Wastes Before Disposal from Nuclear Reactor Facilities," dated December 2, 1985; and Information Notice 88-22, "Disposal of Sludge from Onsite Sewage Treatment Facilities at Nuclear Power Stations," dated May 12, 1988. Licensees should refer to Section 15.11 of Volume 1 of this NUREG report and should contact the NRC staff for further guidance on equipment and solid material releases.

When the LTR was developed, the NRC assumed that decommissioning generally would include the removal of systems and components from onsite buildings before license termination. However, with experience, it has become clear that each licensee uses a different approach for decommissioning, and these approaches are not necessarily consistent with the original assumptions of the LTR. Differences are the result of factors such as (1) the potential for reuse of systems and components, (2) cost of recycling and price of scrap metal and concrete, and (3) cost and availability of disposal options.

It is clear from the LTR technical basis in NUREG-1496, "Generic Environmental Impact Statement in Support of Rulemaking on Radiological Criteria for License Termination of NRC–Licensed Nuclear Facilities," issued July 1997, and NRC draft Regulatory Guide DG-4006, "Demonstrating Compliance with the Radiological Criteria for License Termination," issued July 1998, that the LTR was not intended to apply to releases of "equipment" from the facility. "Equipment" includes anything *not attached to*, or not an integral part of, the building structure. On the other hand, previous guidance (the previous version of Section G.1.1 of NUREG-1757, Vol. 2, Rev. 0) was not prescriptive enough to provide a definitive answer about whether systems and components must be considered "building structures" or "equipment." The previous guidance considered "doors, windows, sinks, lighting fixtures, utility lines, built-in laboratory hoods and benches, and other types of built-in furniture" to be part of the structure. Under that guidance, those items could be included in the FSS and left in place at license termination. It could be argued that, based on the examples provided, many plant systems and components also could be considered "building structures," and, therefore, left in place at license termination. This previous guidance may have been inconsistent with the discussion in the LTR Analysis Commission Paper, SECY-03-0069, which described an expectation that removable materials and equipment would generally not be present at the time of license termination.

The staff has identified a number of acceptable approaches to clarify what building structure-related materials may be left on site at the time of license termination and what criteria should be applied to those materials.

For this discussion only, the NRC staff uses the following descriptions of building structures, systems and components, and equipment:

- "Building structures" include floors, walls, and roofs; components embedded in floors, walls, and roofs (e.g., embedded piping); and items that are attached to and are an integral part of the buildings (e.g., doors and windows).

- "Systems and components" include items attached to a building structure that are not an integral part of the building but provide important functions to the building (e.g., utility lines, sinks, lighting fixtures, built-in laboratory hoods and benches, polar cranes (in power reactors), and major process equipment).

- "Equipment" includes items not attached to the building structure that are generally readily removable from the building. Examples of equipment include furniture or appliances that are not built into or attached to the structure; stocks of chemicals, reagents, metals, and other supplies; motor vehicles; and any other items that normally would not be conveyed with a building when it is sold.

### *G.2.1.3 Building Structures, and Systems and Components that may be left in place at License Termination*

The NRC staff finds the following approaches acceptable to determine what materials may be left in buildings at license termination.

- **Materials Left On Site Meet Previously Approved Release Criteria**—Building structures and systems and components may be left in place if residual radioactivity in all materials is within the licensee's previously approved criteria for releases of solid materials for unrestricted use. Such criteria may have been approved in license conditions, technical specifications, or generic NRC guidance. The criteria could include use of the "no-detect" policy for reactor licensees, or Policy and Guidance Directive FC 83-23, "Termination of Byproduct, Source and Special Nuclear Material Licenses," 1983, for materials licensees (see also Section 15.11 of Volume 1 of this NUREG report for more information about the current approaches to releases of solid materials).

- **Materials Left On Site Meet "Few Millirem per Year"**—Building structures and systems and components may be left in place if residual radioactivity in all materials is volumetrically distributed (not surficial) and if the potential dose from offsite use exposure scenarios is no greater than a few hundredths of a mSv per year (few mrem per year).

- **Materials Left On Site Meet 0.25 mSv/y (25 mrem/y)**—Building structures may be left in place if the potential dose from the residual radioactivity in or on the structures is within the applicable dose criteria of the LTR (for unrestricted use, no greater than 0.25 mSv/y (25 mrem/y) and ALARA).

- **Alternative Approaches**—Licensees also may propose alternative approaches, which the staff will review on a case-by-case basis. Before submitting such alternative approaches, licensees should contact the NRC staff.

For all approaches, the residual radioactivity in building structures, systems and components, and all other media at the site (e.g., soils or groundwater) must be in compliance with the

applicable criteria of the LTR (e.g., for unrestricted use, doses must not exceed 0.25 mSv/y (25 mrem/y) and must be ALARA).

Licensees will perform dose assessments (or use NRC-approved screening dose assessments) to demonstrate compliance with the dose criteria. Typically, licensees may not need to evaluate potential offsite future use exposure scenarios, such as removal of soil for fill material or road base or reuse of concrete as road bed material, because such offsite use exposure scenarios are usually bounded by onsite use exposure scenarios. However, for some of the dose assessments needed for the above approaches, when less conservative and more realistic exposure scenarios are selected, the onsite exposure scenarios may no longer bound potential offsite use exposure scenarios. Thus, in these cases, the licensee should evaluate offsite use exposure scenarios. For additional guidance, see Section I.3.3.3.6 of Appendix I of this volume.

#### *G.2.1.4 Equipment not covered by the LTR*

The LTR does not apply to equipment, so equipment should not be left on the site at license termination. Equipment should be released under the current approaches for releases of solid materials, as discussed in Section 15.11 of Volume 1 of this NUREG report or could be disposed of as radioactive waste.

### G.2.2 Residual Radioactivity Beneath the Surface

The HSA and characterization surveys may indicate that residual radioactivity is present beneath the surface. In the dose modeling, direct exposure, inhalation, and ingestion pathways may be important for residual radioactivity on the surface. However, if the residual radioactivity is located in subsurface soils, additional pathways may become important to dose, and subsurface soil DCGLs may be derived. In these cases, surveys may be conducted, and results may be interpreted in a manner consistent with the dose modeling (see Section G.3 for additional details).

The FSS surveys cracks and crevices in the same manner as other building surfaces, except that these areas should receive judgmental scans when scanning coverage is less than 100 percent.

For painted-over residual radioactivity, the licensee should use the HSA and characterization surveys to determine whether residual radioactivity was fixed in place by being painted over. If so, it may consider the process for its removal in developing the parameters for the dose modeling and may interpret the survey results in a manner consistent with the dose modeling.

### G.2.3 Sewer Systems, Waste Plumbing Systems and Floor Drains

The HSA and characterization surveys determine whether there are unusual or unexpected levels of residual radioactivity in sewer systems and floor drains. Residual radioactivity in sewer systems and floor drains generally does not contribute to the dose pathways in the building occupancy exposure scenario or the residential exposure scenario. Thus, the licensee should calculate the dose from residual radioactivity in sewer pipes using a site-specific exposure scenario and conduct the FSS in a manner consistent with that scenario. If the sewer water is sent to an onsite drainage field or cesspool, the licensee should evaluate and survey any residual radioactivity as subsurface residual radioactivity. If unusual or unexpected results are found during the characterization survey, the licensee should handle the situation on a case-by-case basis.

If sewage is sent to an onsite drainage field, any residual radioactivity is subsurface, and the survey methods discussed in Section G.3.1 are appropriate.

### G.2.4 Ventilation Ducts

The HSA and characterization surveys indicate whether residual radioactivity may be present. External duct surfaces of ventilation ducts are surveyed as if they are a part of the building surface. The licensee should survey internal duct surfaces in a manner consistent with the dose modeling assumptions.

### G.2.5 Piping and Embedded Piping

Embedded piping is piping embedded in a durable material, typically concrete, that cannot be easily removed without significant effort and tools. The HSA and characterization surveys indicate whether residual radioactivity is present in piping. The normal room surveys will adequately account for direct (external gamma) radiation from the pipes when the pipes are in place and undisturbed. The direct (external gamma) dose from the pipes will be in addition to the dose from the residual radioactivity on surfaces in the room. It may also be necessary consider building renovation that would disturb the piping, as described in NUREG/CR-5512, Volume 1, "Residual Radioactive Contamination from Decommissioning," issued October 1992. If this is done, the survey should be consistent with the dose modeling assumptions.

> NRC staff experience has shown that some DPs have not adequately described the methods the licensee plans to use when surveying the embedded piping planned to be left behind. Often, licensees have not discussed the methodology for conducting surveys of embedded pipe planned to be left behind, nor have they provided sufficient justification for the assumptions considered in the dose modeling analysis. Regulatory Issue Summary 2002-02, "Lessons Learned: Related to Recently Submitted Decommissioning Plans and License Termination Plans," issued January 2002, contains a detailed discussion of this issue.

## G.3 Surveys for Special Situations on Land

### G.3.1 Subsurface Residual Radioactivity Surveys

MARSSIM addresses radiological surveys of surface soils only (i.e., subsurface radiological surveys are not within the scope of MARSSIM). Because the MARSSIM FSS method was designed specifically for residual radioactivity in surface soils, if significant amounts of residual radioactivity are located at depth (e.g., significant quantities of residual radioactivity in soils deeper than approximately 15 centimeters), the presence of subsurface residual radioactivity should be taken into consideration in designing the FSS. The licensee should first determine whether it needs surveys of subsurface residual radioactivity. The HSA and other surveys will play an important role in determining whether there is likely to be residual radioactivity in the subsurface. Modeling can also be used to supplement survey data to determine the potential for residual radioactivity to be present in significant quantities in subsurface soils or groundwater due to environmental transport. If the survey data and supplemental modeling indicate that there is little likelihood of significant subsurface residual radioactivity, then subsurface surveys are likely unnecessary.

If the survey data indicates that there is substantial subsurface residual radioactivity, and the licensee plans to terminate the license with some subsurface residual radioactivity in place, the

FSS should consider the subsurface residual radioactivity to demonstrate compliance with the radiological criteria for license termination. To prepare for the FSS, the characterization survey determines the depth of the residual radioactivity. In addition to conventional drilling, the licensee may consider the use of exploratory trenches and pits, where the patterns, locations, and depths are determined using prior survey results or HSA data.

Performing radiological surveys at sites with significant quantities of subsurface residual radioactivity is more complex compared to surveys of surface soils given the relative inaccessibility of the subsurface regions (e.g., subsurface soils cannot be scanned for elevated areas without the extraction of subsurface materials). Additionally, heterogeneous materials are often encountered in the subsurface, and the presence of contaminated groundwater also presents challenges to subsurface radiological surveys (see Appendix F). Because the MARSSIM methodology relies heavily on scanning to identify elevated areas of concern, alternative or supplemental methods are needed when residual radioactivity is present in the subsurface. Modeling may help inform and supplement collection of radiological survey data and help alleviate the challenge of adequately characterizing the subsurface when scanning is not a viable option. NUREG/CR-7021 (A Subsurface Decision Model for Supporting Environmental Compliance) presents a framework focused on development of a conceptual site model referred to as a contamination concern map (CCM). The CCM describes the extent, location, and significance of residual radioactivity relative to the decision criteria. The CCM can be developed with the aid of visualization, geographic information system (GIS), and geostatistical software. As additional data are collected, the CCM transitions from a more qualitative description to a more quantitative and detailed map. Subsurface concentration estimates and uncertainty measures serve as surrogates to scanning to facilitate more optimal sampling designs and decision-making. The approach laid out in NUREG/CR-7026 (Application of Model Abstraction Techniques to Simulate Transport in Soils) presents one potentially acceptable method that may be used in conjunction with radiological survey data to demonstrate compliance. For complex decommissioning cases where subsurface residual radioactivity and groundwater contamination are present, it is important to work with NRC early in the process to discuss acceptable approaches for demonstrating compliance with radiological criteria for license termination.

As discussed above, GIS and geostatistical software are available to assist with designing, performing, and evaluating the results of radiological investigations. GIS tools can be used to help with creation of conceptual models (e.g., provides spatial context and a better understanding of site features that may control or enhance radionuclide transport in the environment). Figures created with GIS software can also assist with identifying relatively homogeneous areas of residual radioactivity for delineation of survey units. Examples of features that can be captured on a figure using GIS tools include the following:

- Study area and property boundary
- Buildings where residual radioactivity may be present
- Roads
- Surface water features (streams, ponds, runoff basins, ditches, culverts)
- Underground features (underground storage tanks, piping)
- Topography, surface geology, and outcrop locations
- Hydrostratigraphic surfaces and isopach maps

- Water table and potentiometric surfaces
- Sampling locations
- Monitoring well locations
- Contaminant distributions

Geostatistical tools can be used to create figures showing contaminant distributions, predict radionuclide concentrations in areas where no data exists, and identify areas with a higher probability of residual radioactivity above levels of concern. This information can be beneficial in designing the scoping, characterization, and remediation surveys to define the nature and extent of residual radioactivity (e.g., optimizing the number and locations of samples). Figure G-1 presents an example use of geostatistical software.

If subsurface residual radioactivity is present, dose modeling may be conducted for both surface (if present) and subsurface soils and DCGLs developed for each. In these cases, the MARSSIM methodology will need to be supplemented or an alternative methodology will need to be developed to demonstrate compliance with radiological criteria for license termination (MARSSIM only addresses residual radioactivity at the surface). Because the depth and thickness of residual radioactivity are correlated to dose, the modeling should reflect the actual distribution of radioactivity in the survey unit. For example, for certain radionuclides (e.g., those whose risk is dominated by the plant ingestion pathway), the thickness of residual radioactivity is strongly correlated to dose. If the modeling assumes a thinner layer of residual radioactivity than is present, then the risk could be significantly underestimated. If the modeling assumes a thicker layer of residual radioactivity than is present, then the risk could be significantly overestimated. Additionally, for some radionuclides (e.g., those whose risk is dominated by the external dose pathway), the surface concentration may drive the risk as radiation emitted from residual radioactivity located at greater depth may be attenuated in the soil column and not contribute to dose. Therefore, if vertical heterogeneity is an issue, then it may be necessary to take discrete samples to ensure that higher concentration residual radioactivity at the surface is not diluted in cleaner materials at depth. Dose modeling can be used to determine the sensitivity of dose to these parameters, and the soil sampling design should ultimately be consistent with the modeling used to develop the DCGLs. Ideally, sufficient resolution in the sampling data would be available to evaluate vertical heterogeneity and calculate appropriate concentrations for comparison against DCGLs derived for specific depths and thicknesses and/or for the total thickness of residual radioactivity to ensure dose is not underestimated.

GIS tools can be used to assist with designing and interpreting results of radiological surveys. Figure G-1 shows a map that includes the location of two hypothetical tanks (a). Leaks are known to have occurred near the tanks. GIS information on the location of important features and topography of surficial or subsurface structure can be used to identify areas where residual radioactivity may be present and more likely to have been transported (e.g., surface runoff direction). GIS information and geostatistical tools can be helpful in designing survey plans and identifying areas most likely to be above risk-based thresholds. For example, the geostatistical tools available in codes such as Visual Sample Plan and Spatial Analysis and Decision Assistance (SADA) can be used to analyze data and extrapolate data in areas where no data is available. Figure G-1 illustrates the use of SADA (Version 5) in creating a display of the results of sampling (b) and a 3D visualization of the volume of soil most likely to be impacted based on the sampling results and use of geostatistical tools available in the code (c).

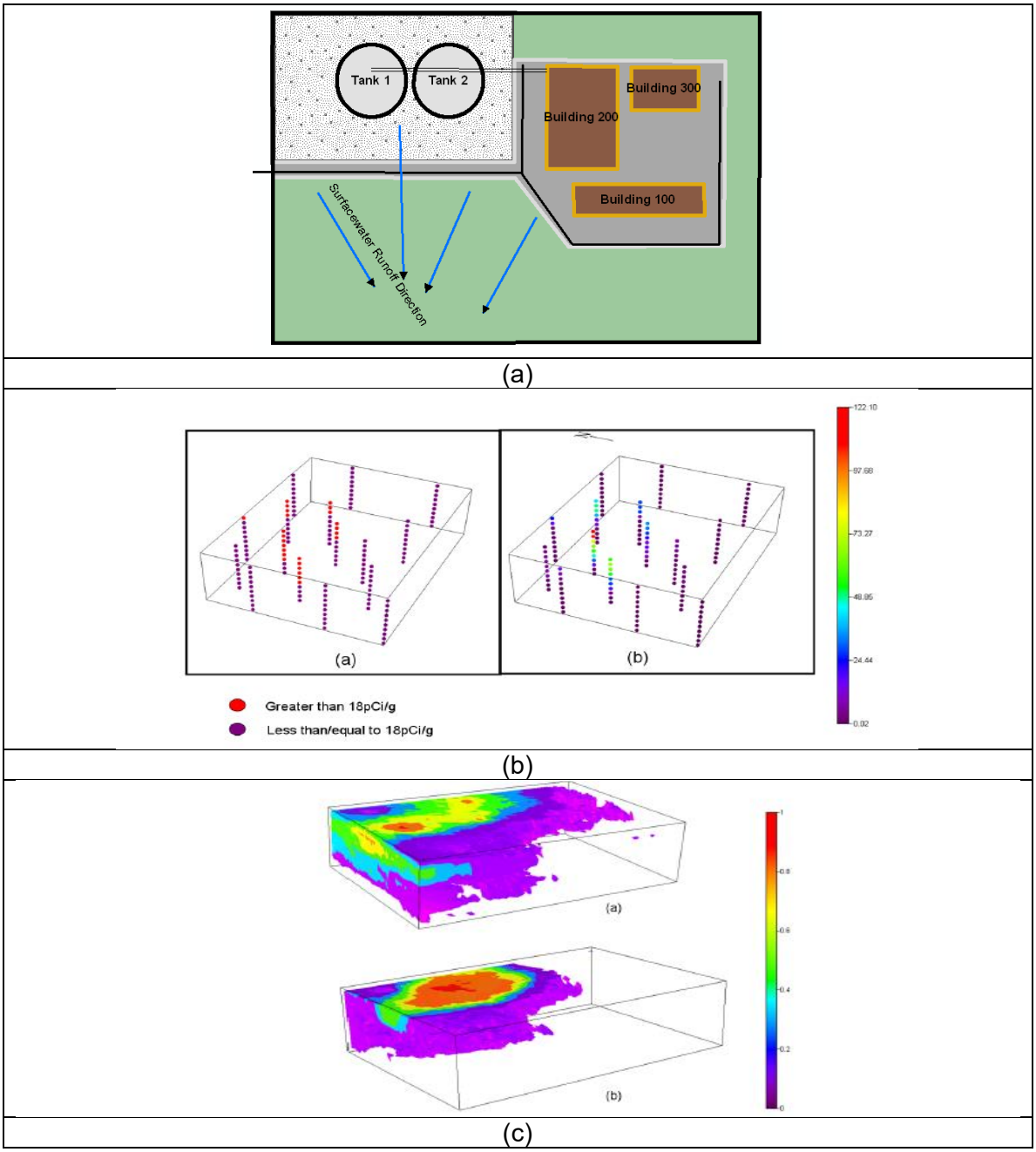


**Figure G.1 Use of GIS and Geostatistical Tools** *(a) to Identify Potential Areas where Residual Radioactivity could be Present, (b) to Visualize Borehole Sampling Results and (c) Interpolate Data and Determine Probability of Exceeding a Threshold following Scoping (top) and Create a more Refined Map following Characterization (bottom)*

When the licensee establishes the appropriate DCGLs and mixing volumes, based on an acceptable site-specific dose assessment, the FSS takes core samples to the measured depth of the residual radioactivity. The number of cores to be taken is initially the number (*N*) required

for the WRS or Sign test, as appropriate. The adjustment to the grid spacing for an EMC is more complicated than for surface soils, because scanning is not applicable. The core samples should be homogenized over a soil thickness that is consistent with assumptions made in the dose assessment, typically not exceeding 1 meter in depth. It is not acceptable to average radionuclide concentrations over an arbitrary soil thickness. The appropriate test (WRS or Sign) is then applied to the sample results. Triangular grids are recommended, because they are slightly more effective in locating areas of elevated concentrations. Site-specific EMCs may also need to be developed to demonstrate regulatory compliance and should take into consideration key radionuclides, pathways, and exposure scenarios important to dose. For subsurface residual radioactivity at depth, the groundwater pathway and total inventory may drive the risk (i.e., small elevated areas of concentration may not be important to dose). Most intrusion scenarios assume some minimum degree of mixing of excavated soils; therefore, mixing arguments can be presented when determining the minimum volume of soil of interest in developing EMCs. The NRC has not yet developed generic guidance for performing an EMC for subsurface samples; therefore, licensees should discuss this matter with the NRC staff on a case-by-case basis.

The sampling approach described above may not be necessary if sufficient data to characterize the subsurface residual radioactivity are available from other sources. For example, for some burials conducted under prior NRC regulations, the records on the material buried may be sufficient to demonstrate compliance with the radiological criteria for license termination.

### G.3.2 Surveys of Excavations and Use of Backfill Soil for Excavated Land Areas

#### *G.3.2.1 Surveys of Excavations*

In cases where a licensee must remediate a site through excavation of subsurface residual radioactivity above clean-up levels, several options are available to the licensee to demonstrate compliance with radiological criteria for license termination. Although a backfilled excavation represents the final configuration of the site, it is a reasonable expectation that the licensee will perform the FSS on the open excavation prior to backfilling, if the survey can be performed safely. This is due to the potential cost and difficulty associated with adequately sampling a backfilled survey unit and the fact that scanning the entire depth of backfill would likely not be possible for most situations. Sampling and scanning of the open excavation are also beneficial by helping to ensure that residual radioactivity above levels that would lead to an exceedance of the dose criteria are removed and appropriately disposed. When a FSS is performed on an open excavation, it is important to document the locations and depth range below final grade represented by sampling as well as the general topographical layout of the excavation relative to final grade to understand the final distribution of residual radioactivity at the site and to facilitate comparison to release criteria. Additionally, it is important for the licensee to communicate with the NRC staff to allow the NRC to perform a confirmatory survey that provides independent evaluation of radiological conditions before the licensee backfills the excavation.

Survey instruments should be used that are appropriate for evaluating the radioactive contamination of interest and all accessible surfaces of Class 1 survey units should be evaluated. Specialized equipment such as extended instrument cables/poles, trench boxes, or other tools and equipment may be needed to perform surface scans and to obtain samples from locations within the excavation and sidewalls suspected of contamination. Deep excavations may require significant preparatory work both to perform the excavation and to make an excavation safe to access, examples of this include pumping to lower the water table, use of

steel pilings to shore up excavation side walls, and appropriate respiratory protection if toxic chemicals are present or could displace oxygen. In all cases, the health and safety of workers and the public should take priority.[1]

There are several approaches a licensee can take for sampling an excavation. Although MARSSIM does not apply to subsurface soils, a MARSSIM-based approach may be extended to subsurface problems. The survey classification of an excavation should consider whether the entire excavated area, including the floor and the sidewalls, has the same contamination potential. Also, if remediation of soil took place to meet the release criteria, then a MARSSIM Class 1 classification would be appropriate. In this case, if the floor and sidewalls of the excavation both possess the same potential for contamination, then the same level of survey effort should be applied to both areas. However, there may be cases where sloped sidewalls into an excavation are created to provide safe access for remediation activities, and the contamination potential of the sidewalls could be lower than the bottom of the excavation. In this example, the new sidewalls may be a Class 2 buffer area, whereas the floor of the excavation is a Class 1 area. As such, it is important to evaluate excavation practices and sampling strategies during the DQO process to ensure that classification and survey methodologies account for actual site conditions. It is also important to consider the potential for ongoing decommissioning activities to contaminate buffer areas in an excavation. Although the sidewalls or bottom of the excavation may not be planar surfaces, sampling locations could be established based on a flat, final graded area (see Figures G-2 (a) and (b)). The depth of sampling of the excavation should be considered during the DQO process. Sampling of sloped excavation sidewalls that are excavated to a depth shallower than the bottom of the excavation pose additional considerations due to their difference in elevation below grade. Different sample collection strategies such as sampling more horizontally or vertically into the sidewall, may also be considered during the DQO process. In addition, a sampling strategy may need to take into account the slope of sidewalls, sample density, thickness of strata, and multiple DCGLs developed to represent different soil intervals. Licensees are encouraged to discuss with the NRC staff the appropriate level of detail and technical features of their sampling strategy.

In many cases, licensees have used a layered approach in which multiple subsurface layers or strata are considered individually and then the cumulative risk from the multiple layers or strata are assessed. If the sidewalls of the excavation are considered the same Class 1 survey unit consistent with the bottom of the excavation, then the cumulative risk from all of the strata (below and above the bottom of the excavation) should be considered. Alternative approaches are available to assess the data.

As noted above, in some cases multiple sets of DCGLs may have been approved for different strata or layers below final grade based on dose modeling. For example, surface soil is important to the external dose and inhalation pathways, intermediate depth soil corresponding to the till depth is potentially important to the plant ingestion pathway, and subsurface soil may be important to the groundwater pathway. Intrusion scenarios are considered to develop subsurface DCGLs for otherwise inaccessible subsurface soils at depth (see Appendix J). In cases where different sets of DCGLs are developed for different strata, it is important to ensure the average contaminant concentration in each designated stratum is lower than the applicable

[1] Proper excavation safety practices should always be applied in accordance with Occupational Safety and Health Administration (OSHA), such as the use of support systems, and/or sloping and benching, to stabilize the excavation site. Refer to the US Army Corps of Engineers, Safety and Health Requirements Manual EM 385-1-1 for additional details.

DCGL and that any elevated areas are appropriately investigated and addressed. A sum of fractions approach can be used to assess the cumulative risk associated with multiple strata. For example, a sum of fractions approach may entail calculating the sum of fractions for each sampling location considering the entire soil column, and then calculating the average sum of fractions for all the locations if the sum of fractions is greater than 1 for any one location, or evaluating the sum of fractions for multiple strata using the average concentration for each strata. In all cases, it is important to understand the basis for development of the DCGLs (thickness, depth, distribution, and area of residual radioactivity) for each stratum to ensure that risk is not underestimated. It is also important to understand the dose modeling assumptions with respect to the reuse of soils in the excavation (i.e., use of clean or slightly contaminated soils) to ensure the risk is not underestimated. In complex cases involving subsurface residual radioactivity, it is always prudent to calculate the final estimated dose for the compliance scenario(s) based on the final configuration and measured radionuclide concentrations of residual radioactivity at the site through dose modeling.[2]

As stated above, it is important to adequately assess the radiological characteristics of residual radioactivity in soils used for backfilling an excavation. For each of the strata for which DCGLs are developed, radioanalytical results of backfill soil sampling should be analyzed and appropriately included in the sum of fractions calculations discussed in the preceding paragraph. In some cases, DCGLs developed for strata below the excavation bottom could be significantly greater than DCGLs developed for more shallow strata (i.e., strata associated with backfill soil). Therefore, it is important to ensure that the backfill soil concentrations are compared to the appropriate DCGL to ensure that the risk is not underestimated. In many cases, the backfill soil is sampled after the soil is excavated, in addition to the surveys conducted before the soil is emplaced in the excavation.

In some cases, it may be appropriate and more straightforward to derive a single set of DCGLs based on the entire thickness of residual radioactivity, rather than develop multiple sets of DCGLs for different strata thicknesses. However, sensitivity analyses may need to be conducted to ensure that the risk is not underestimated using this approach. For example, for radionuclides dominated by the external dose pathway, a similar DCGL may be developed for a small thickness of residual radioactivity (e.g., 15 cm) compared to a larger thickness of residual radioactivity, as residual radioactivity at depth may not contribute to the external dose pathway due to attenuation in the soil column. The concentration of the surface layer would then be appropriate to compare to the DCGL. If the plant pathway is important, soil concentrations down to 1 m may be important to dose (depending on the plant types and depth of roots). When a mix of radionuclides are present, certain radionuclides may be transported to greater depths than others. DCGLs derived for all of the radionuclides based on the depth of residual radioactivity for one radionuclide may lead to underestimate of risk for other radionuclides. In these cases, care should be taken to ensure that residual radioactivity in the subsurface is adequately characterized to determine the lateral and vertical extent of contamination and elevated concentrations are appropriately considered (i.e., depth discrete surveys or sampling may be needed to capture heterogeneity in soil concentrations and ensure that elevated areas above DCGLs are not diluted by averaging concentrations with clean soils).

[2] Although commonly used codes such as RESRAD-ONSITE only consider one average soil concentration as input to the code, the code can be run multiple times and the doses summed to assess the contributions of multiple strata.

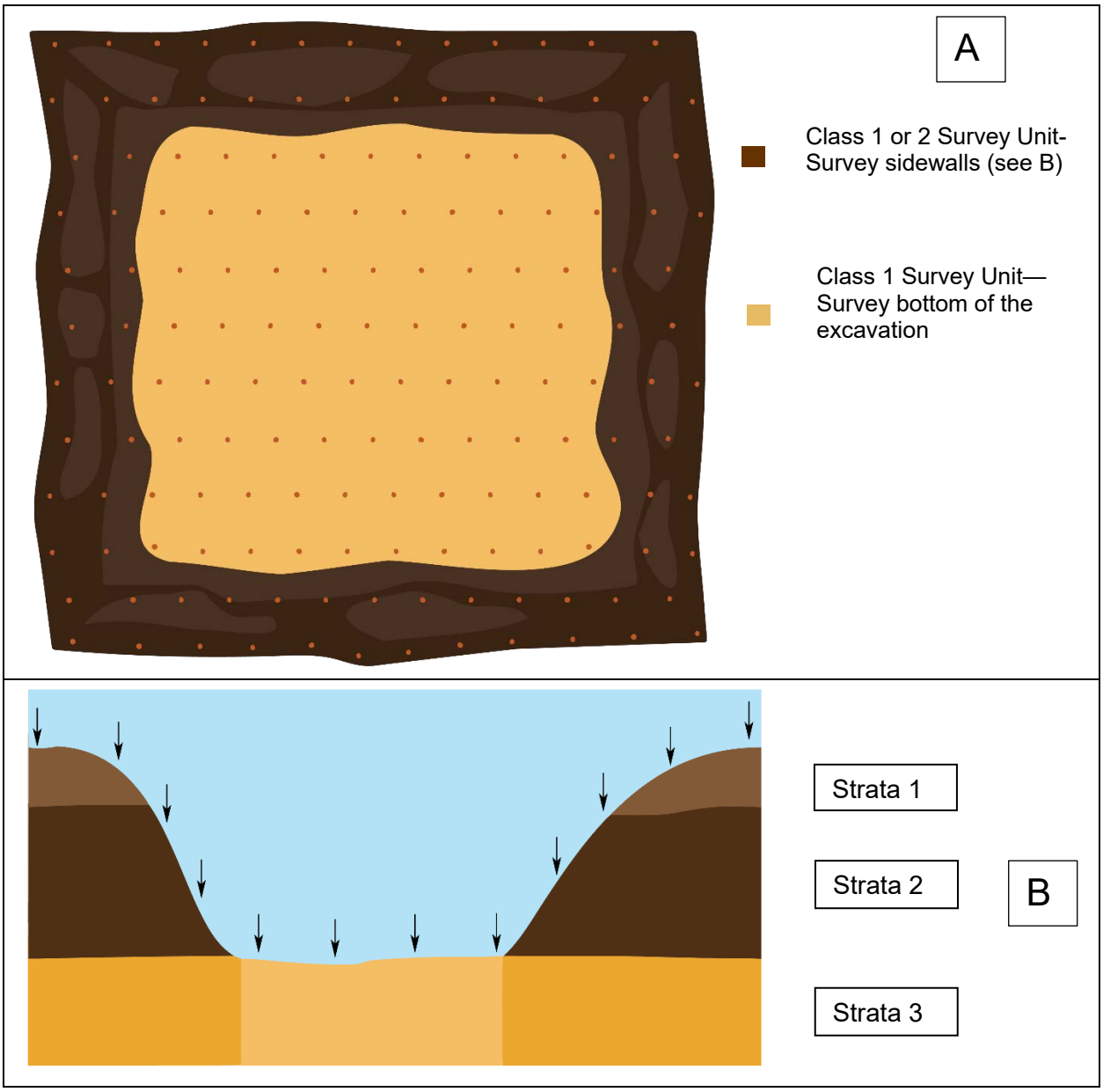


**Figure G.2 Sampling Strategies for Excavations** (*A) Use of MARSSIM Survey Design Approach for Excavation Bottom and Sidewalls [Plan View Map on top] and (B) Depiction of Potential Options for Sampling into Excavation Sidewalls to Assess Residual Radioactivity in Remaining Soils (Various Strata for which DCGLs may be Derived are Depicted) [Cross-Section Map on bottom]*

Excavated land areas at sites undergoing decommissioning will often require backfill soils. Multiple options exist for backfill sources, such as offsite areas, nonimpacted areas on site, or impacted areas on site that have been appropriately surveyed.

### *G.3.2.2. Backfill from Nonimpacted Onsite and Offsite Areas*

Licensees have typically proposed to use backfill from non-impacted areas onsite or from offsite locations. If the licensee is assuming there is no added residual radioactivity in the backfill, an analysis should be performed to support this assumption (i.e., that the backfill soils do not contain residual radioactivity). Residual radioactivity, as defined in this NUREG series, includes radioactivity from all licensed and unlicensed sources used by the licensee, but excludes background radiation. If there is uncertainty that backfill soils are non-impacted, one potential method to support this assumption of no added residual radioactivity would be to use a two sample statistical test such as a Scenario B type analysis to show indistinguishability from background, as described in Chapter 13 of NUREG–1505. See Section G.6 for more information about Scenario B. If Scenario B is used, the licensee should discuss appropriate values for the LBGR, and UBGR (or width of the gray region) with the NRC. Although some form of radiological survey is expected, other approaches proposed by the licensee may also be acceptable.

### *G.3.2.3. Backfill from Impacted Onsite Areas*

Re-use of soils from radiologically impacted areas as backfill at a site undergoing decommissioning has been allowed by the NRC on a case-by-case basis. Licensees should continue to discuss proposed soil re-use plans with the NRC, as there are potentially complex issues associated with radiological measurement capabilities and site-specific dose assessments. The following guidance may assist licensees in developing re-use plans, though site-specific conditions may lead to additional issues for consideration.

If licensees plan to re-use soil from impacted areas onsite, characterization and radiological surveys should be completed to the rigor of a FSS. The MARSSIM survey unit classification (i.e., Class 1, 2, or 3) should be considered to ensure that the number of measurements and scanning coverage are adequate. Licensees should also consider the depth at which surface soils can be measured via surface scanning and adjust excavation methodologies accordingly (e.g., a Class 1 area should receive 100% surface scanning followed by systematic soil sampling). In this case, if the entire depth of re-use soil cannot be adequately surveyed it may be necessary to excavate and survey soil via lift depths consistent with surface soil dose modeling and instrument capabilities. Similarly, licensees should ensure that the sampling and scanning performed for Class 2 and 3 areas satisfy the survey objectives. Additionally, the effects of re-using soil onsite should be evaluated in the context of the final site dose assessment. The licensee should consider the final configuration of reused soil and may take credit for clean cover materials in estimating dose for residual radioactivity remaining at the site. The reuse of impacted soils, depending on the methods used, may include intentional mixing of soils containing residual radioactivity. Licensees should refer to Volume 1, Section 15.13, of this NUREG for additional guidance on intentional mixing of soils to meet license termination criteria. It is recommended that the licensee fully consider the implications of re-using impacted soils during the planning and DQO processes and discuss plans with the NRC as needed. Additional details regarding radiological surveys for reused soils are discussed in the following paragraphs.

Excavated soils may be segregated at a site, based on the level of residual radioactivity for potential reuse (e.g., impacted soil may be stockpiled for possible reuse as backfill or grading materials on a site). If the residual radioactivity includes radionuclides that can be measured in a scan survey (i.e., contaminants are not hard-to-detect radionuclides that cannot be detected in

a scan survey[3]), then the licensee could use scanning as a first method to evaluate the soil. If scanning indicates that the soil contains residual radioactivity above background when it is expected to be free of residual radioactivity, or above release criteria when DCGLs are developed for reused soils, then the licensee should determine whether it is acceptable to reuse the soil at the site. In some cases, the soils may not be suitable for stockpiling and reuse[4]. The stockpiling of soil should be limited so that only scanned strata are excavated before the newly exposed soil is scanned. The thickness of the excavated lifts should be consistent with the scanning instrument capability and scanning coverage consistent with the survey unit classification. Alternatively, soils can be excavated, transported to a suitable laydown area, and then scanned in the laydown area similar to a FSS prior to stockpiling. In these cases, the thickness of soil in the laydown area for scanning should be consistent with the scanning instrument capability and survey unit classification (e.g., in a Class 1 area, the areal coverage should be 100 percent and the survey instrument should be able to detect residual radioactivity at the bottom of a soil layer in the laydown area). Stockpiled soil and materials from impacted areas that are heterogeneous may result in local areas of increased radioactivity levels. In these situations, it may be useful to rescan the soil and materials after emplacement to determine whether additional evaluations or spot remediation is warranted, especially if the soils are used for final grade material.

In some cases, licensees have used conveyor belts to scan excavated materials. Soil characteristics such as soil type, density, and moisture content, and measurement sensitivity are factors to consider in the DQO process when using these systems.

In summary, when stockpiled soils are planned for use as backfill or grading material and remain onsite at the time of license termination, it is important to develop a survey plan that takes into account the characteristics of the soil, measurement methods to be used before and after emplacement, and applicable radiological criteria, which may be based on site-specific dose modeling.

Although DCGLs may have been developed to guide site remediation activities, it is also prudent to perform dose modeling using the final configuration and concentrations of residual radioactivity at the time of license termination.

### G.3.3 Rubble, Debris, and Rocks

Rubble, debris, and rocks can include naturally occurring rocks (either in place or in piles), pieces of concrete or rubble from buildings that have been razed, sheet metal disposed of as trash, asphalt, fly ash, and similar material. The HSA and characterization surveys determine the volumetric extent and residual radioactivity concentration. If the materials are contaminated, they would be disposed of as radioactive waste. If the radioactivity is not substantially elevated, the licensee may evaluate the rubble, debris, and rocks as part of a larger survey unit. When these materials will be evaluated as part of a larger survey unit and when they are found on a relatively small fraction of the area of a survey unit, the volumetric soil DCGL should be used uniformly throughout the survey unit. However, the licensee should justify the reasonableness of modeling rocks and rubble as soil.

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3 See Appendix A of this document and Section 4.3.2 of MARSSIM, Revision 1 for additional information on use of surrogate radionuclides for hard-to-detect radionuclides.

4 Soil that is deemed unacceptable for reuse, may be disposed of in accordance with 10 CFR Part 20, Subpart K.

### G.3.4 Paved Parking Lots, Roads, and Other Paved Areas

The HSA and characterization surveys determine whether the residual radioactivity is on or near the surface of the paving and whether there are significant concentrations of residual radioactivity beneath the paving. If the residual radioactivity is primarily on top of the paving, then the licensee should take the measurements as if the area were normal soil. Depending on how large the paved area is, the licensee may include it as part of a larger survey unit, or it may be its own survey unit. If the residual radioactivity is primarily beneath the paving, the licensee should survey it as subsurface residual radioactivity, as discussed above.

### G.3.5 Recontamination

Finally, measures should be employed to avoid recontamination of previously remediated areas and the need for additional surveys or analyses to demonstrate compliance with release criteria. Recontamination of cleaned survey units or areas on site can occur particularly at decommissioning sites where (i) decommissioning is conducted in phases, (ii) partial site releases are conducted, and (iii) remediation occurs over long time-frames. Contamination of cleaned areas from residual radioactivity in unremediated areas can occur due to environmental and human-induced transport such as through surface water runoff, air transport; and soil moving or demolition activities. Should NRC staff suspect recontamination of previously surveyed areas, additional surveys, dose modeling, or remediation may be necessary to demonstrate compliance with release criteria.

## G.4 Surveys Associated with Multiple Radionuclides and Media

In cases where residual radioactivity is present in multiple media and multiple radionuclides are present, it may be necessary to adjust the dose modeling approach used to derive DCGLs, as well as the survey design approach used to demonstrate compliance with radiological criteria for license termination to account for the cumulative dose from multiple radionuclides and media. For example, consider a site with residual radioactivity present on the surfaces of a building, surface and subsurface soils, and nearby stream bed sediments. Multiple key radionuclides contributing significantly to the dose are present in each of the media and different key radionuclides are important for each of the media. Given the previous industrial use of the site and presence of residential areas in close proximity to the site, reasonably foreseeable future land use scenarios include industrial and residential use of the site within 100 years of license termination. Exposure to stream bed sediments may also occur due to recreational use of the stream such as swimming, boating, and fishing. Therefore, a recreational scenario may also be considered for the streambed sediments. DCGLs are typically derived for each of the contaminated media based on the release criteria (e.g., 0.25 mSv/y [25 mrem/y] for unrestricted release). However, given limitations in the dose modeling codes used to derive the DCGLs, receptors are typically assumed to be exposed to just one of each of the media. Likewise, given use of different DCGLs for each of the different media, the survey design does not automatically consider the contributions of multiple media in demonstrating compliance with release criteria.

With respect to consideration of multiple radionuclides, the sum of fractions approach can be used to account for multiple radionuclides in the same survey unit. Chapter 11 in NUREG-1505 (NRC, 1998) provides guidance and an example on how to use the sum of fractions approach to design the survey to determine the appropriate number of samples to demonstrate compliance with release criteria while ensuring decision criteria are met (e.g., acceptable Type I [false positive] and Type II [false negative] errors). However, this example is only applicable to a single survey unit type (e.g., soil, building, streambed sediments).

While surface and subsurface soils may have separate DCGLs, similar to the approach used for multiple radionuclides, surface and subsurface soils can be considered together when demonstrating compliance because they can be assumed to be located in the same survey unit(s). However, soil survey units are typically separate from building survey units and streambed sediment survey units. Therefore, if the demonstration of compliance is made for each media survey unit type based on clean-up levels derived at the compliance limit, if a member of the public could be exposed to multiple media and survey unit types in the same year, the annual release limit could be exceeded.

Licensees can take a conservative approach to addressing cumulative dose associated with each of the media by re-calculating the DCGLs for each of the media assuming a portion of the compliance limit (e.g., 45 percent of the 0.25 mSv/y [25 mrem/y] unrestricted release limit for (i) the soils, and (ii) building; and 10 percent for (iii) streambed sediments). It is important to note that re-calculation of the DCGL is necessary prior to designing the survey to ensure that the survey design is based on the correct width of the gray region (e.g., additional samples may be needed if the width of the gray region is smaller due to the lower DCGL or UBGR used in the survey design). While assuming both a residential and industrial scenario occur at the same site may be overly conservative with occupancy factors approaching 100 percent or greater, this would be an acceptable approach to consideration of cumulative dose associated with all sources/media and radionuclides.

Similar to methods used to consider elevated areas or "hot spots" (see Appendix I), the licensee may also develop DCGLs based on more reasonable exposure assumptions if a receptor is assumed to be exposed to multiple sources or media. For example, if the dose contributions from both the building and the soils are considered in demonstrating compliance, occupancy factors may be adjusted lower to account for the time spent in the building and time spent outside in an industrial scenario when developing DCGLs for the building and soils. Codes such as RESRAD-BUILD and RESRAD-ONSITE allow for occupancy factor parameter values to be adjusted such that the total time fraction is not greater than 1 (e.g., the residential farmer scenario assumes time fractions of 0.12 (outdoors) and 0.66 (indoors), and building occupancy scenario assumes a time fraction of 0.25 for a total time fraction greater than 1.0) . Other acceptable approaches for adding realism to the dose calculations may be available and should be discussed with the NRC staff early in the process.

## G.5 Consideration of Elevated Areas in Survey Designs

Using the FSS approach, sites are typically broken up into a number of survey units without regard to the presence of elevated areas. In cases where multiple elevated areas exist in relatively close proximity to one another, the licensee should consider restructuring the survey units so that the elevated areas are evaluated in as few survey units as practical and which meet the guidance for survey unit size and have a shape similar to typical local property lots for housing or industry.[5] This may be unnecessary if physical barriers exist such that it would be unlikely that future land use would result in a single occupant having potential exposure from all elevated areas. Alternatively, if the elevated areas do not contribute significantly to the potential exposure, it may be possible to simply show that the contributions of the elevated areas do not exceed the criteria for any of the adjoining survey units when considered as an additional source of exposure.

[5] Novel dose modeling approaches to addressing potential gross over-estimations in dose when considering multiple elevated areas is found in Appendix I.

With regard to delineation of elevated areas, sampling data and scanning data are typically used to define the boundaries of the elevated area. When defining the boundaries of the elevated area, the area should not be intentionally established to incorporate multiple sample results that are significantly less than the DCGLs and which would, therefore, intentionally lower the average activity level of the elevated area. In cases where the survey unit boundaries split the elevated area, the entirety of the elevated area should be considered as being in the confines of both survey units or else the survey units should be reconfigured so that it exists in its entirety in just one survey unit. Other approaches may be proposed by the licensee that may be acceptable and should be discussed with the NRC staff early in the process.

When elevated areas are encountered, good housekeeping practices consistent with ALARA criteria should be considered. For example, if it is practical or cost-beneficial to remove a small area of elevated activity (e.g., a couple buckets of contaminated soil or caulk in a floor seam), removal activities would likely be considered consistent with ALARA criteria. Minor removal activities would also help alleviate complications associated with consideration of elevated area contributions to dose and interpretation of anomalous survey results.

## G.6 Additional Information and Example on Conducting Scenario B

As noted in Section 2.4, the default assumption for FSSs is that the survey unit is considered contaminated above the limit (i.e., the null hypothesis is that the concentrations of residual radioactivity exceed the DCGLs). This assumption and null hypothesis are considered Scenario A. Typically, statistical tests are used to demonstrate that the average or median of the measurements in the survey unit are not above the DCGL, with the burden of proof on the licensee to show that the survey unit meets the release criteria. In special cases (i.e., exceptions) it may be appropriate to assume an alternative scenario in which the survey unit is considered clean (i.e., the null hypothesis is that the concentrations of residual radioactivity meet the release criterion). This alternative, Scenario B, should typically only be used when residual radioactivity in the survey unit is within the range of measurement and/or background variability, making it difficult to distinguish between the residual radioactivity and background. For example, sites with highly variable background and comparatively small DCGLs may desire to demonstrate that measurements in the survey unit are indistinguishable from measurements in reference areas. Other Scenario B applications may include the use of gross alpha or gross beta measurements, or nuclide-specific measurements when the nuclide is in background, similar to Scenario A. Licensees considering the use of Scenario B for compliance with 10 CFR 20 Subpart E are strongly encouraged to contact with the NRC staff early in the planning process. In all cases, the licensee should discuss its plans with the NRC to determine the acceptability of using Scenario B, as well as determining appropriate values for the test parameters.

As a Scenario B example, this section describes five major steps to demonstrate indistinguishability from background when the Wilcoxon Rank Sum and Quantile tests are used: (1) assess background variability using the Kruskal-Wallis test; (2) determine a concentration of radioactivity that is indistinguishable from background; (3) perform the Wilcoxon Rank Sum test; (4) perform the Quantile test if the survey unit “passes” the Wilcoxon Rank Sum test; and (5) perform the elevated measurement comparison test. Information on these methods is contained in Chapters 6, 7 and 13 of NUREG-1505, and summarized below.

Prior to performing statistical tests, measurements are made in several different background reference areas. The number of samples needed depends on the probability of Type I error (α) and Type II error (β) that are considered acceptable. The null hypothesis is that there is no significant variability between different background reference areas, so a Type I error would incorrectly conclude there is a significant difference between the background reference areas when there is no significant difference. Using the DQO process, a value of α is selected for an acceptable frequency of a Type I error rate. Table 13.5 of NUREG-1505 provides results of calculations of the power of the F-test (parametric complement to the nonparametric Kruskal-Wallis test) as a function of the number of reference areas and the number of measurements in each reference area for α values of 0.05, 0.10 and 0.20. Based on the results of these calculations, NUREG-1505, Chapter 13, indicates that 4 reference areas with between 10 and 20 measurements in each reference area should generally be adequate, and that an α value of 0.10 is a reasonable default value. Other α values, number of reference areas and number of measurements in each reference area could be found to be reasonable during the DQO process. For planning purposes, it may be convenient to select multiples of five measurements for performing the subsequent Quantile test because Table A.7a of NUREG-1505 contains values as multiples of five survey measurements.

In this Scenario B example, measurement data from 10 measurements in 4 reference areas are provided in Table G-1. The Kruskal-Wallis test is performed on this data to determine if significant variability exists in the background reference areas. The measurements for the reference areas are pooled and ranked, and then the sum of the ranks for the individual reference areas ($R_i$) and the mean measurement are determined. A test statistic (K) is obtained using NUREG-1505 equation 13-3 (see equation below), where *N* is the total number of measurements in all the reference areas *i*=1 to k reference areas; $n_i$ is the number of measurements in a given reference area; and $R_i$ is the sum of the ranks of the measurements in a given reference area:

$$K = \frac{12}{N(N+1)}\left(\sum_{i=1}^{k}\frac{R_i^2}{n_i}\right) - 3(N+1)$$

The value of K is then compared with the critical value ($K_c$) found in Table 13.1 of NUREG-1505. If K is greater than $K_c$, the null hypothesis is rejected, and it is concluded that there is significant variability between the reference areas. For the example data in Table G-1, the K value of 14.0 is compared to $K_c$ for three reference areas (four reference areas minus one reference area). The value of $K_c$ is lower than K, which ranges from 11.3 (for an α value of 0.01) to 4.6 (for an α value of 0.2). Therefore, the null hypothesis can be rejected with high confidence. It is concluded that there is significant variability between the reference areas, which helps justify use of Scenario B for determining indistinguishability from background.

As mentioned above, an alternative to the Kruskal-Wallis test is the F-test to determine if variability between the means of potential reference areas is statistically significant. Chapter 13 of NUREG-1505 contains information on the use of the F-test for this purpose. Although the Kruskal-Wallis test (or F-test) is used to determine if it is appropriate to consider reference area variability in applying Scenario B, it may be acceptable to not conduct either test if it is decided that there are significant differences among the potential reference areas for a survey unit. Thus, if it is considered appropriate to give background variability the benefit of the doubt, the Kruskal-Wallis and F-test test need not be conducted. Not conducting these tests is essentially the same as setting α equal to 1.0.

**Table G-1 Calculation of $\omega^2$ for the Example Data**

| | Measurements | | | | Measurement Ranks | | | | Measurements Squared | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | Area 1 | Area 2 | Area 3 | Area 4 | Area 1 | Area 2 | Area 3 | Area 4 | Area 1 | Area 2 | Area 3 | Area 4 |
| 1 | 0.27 | 1.04 | 2.45 | 3.77 | 6 | 13 | 27 | 39 | 0.07 | 1.08 | 6.00 | 14.21 |
| 2 | 1.87 | 0.39 | 0.34 | 2.63 | 20 | 9 | 8 | 31 | 3.50 | 0.15 | 0.12 | 6.92 |
| 3 | 0.97 | 2.07 | 3.06 | 4.05 | 10 | 23 | 37 | 40 | 0.94 | 4.28 | 9.36 | 16.40 |
| 4 | 1.01 | -0.57 | 2.83 | 1.72 | 11 | 2 | 35 | 19 | 1.02 | 0.32 | 8.01 | 2.96 |
| 5 | 2.08 | 1.97 | 1.09 | 1.50 | 24 | 21 | 14 | 17 | 4.33 | 3.88 | 1.19 | 2.25 |
| 6 | 1.62 | -0.22 | 0.26 | 2.47 | 18 | 3 | 5 | 29 | 2.62 | 0.05 | 0.07 | 6.10 |
| 7 | 0.30 | 1.39 | 2.80 | 1.42 | 7 | 15 | 34 | 16 | 0.09 | 1.93 | 7.84 | 2.02 |
| 8 | 1.98 | 0.05 | 2.77 | 2.47 | 22 | 4 | 33 | 28 | 3.92 | 0.00 | 7.67 | 6.10 |
| 9 | 2.18 | -0.75 | . 2.42 | 2.76 | 25 | 1 | 26 | 32 | 4.75 | 0.56 | 5.86 | 7.62 |
| 10 | 1.02 | 2.50 | 2.86 | 3.35 | 12 | 30 | 36 | 38 | 1.04 | 6.25 | 8.18 | 11.22 |
| sum | 13.30 | 7.87 | 20.88 | 26.14 | 155 | 121 | 255 | 289 | 22.28 | 18.50 | 54.30 | 75.80 |
| average | 1.33 | 0.79 | 2.09 | 2.61 | | | | | | | | |
| ave: sqd | 1.77 | 0.62 | 4.36 | 6.83 | | | | | | | | |

The next step is to determine the LBGR. The LBGR is the concentration level above background that may be considered distinguishable from background and can be established using a measure of the variability among the background reference areas. The mean square between reference areas, $s_b^2$, and mean square within reference areas, $s_w^2$, are used to calculate the component of the variance ($\omega^2$). These test parameters can be calculated using methods described in Chapter 13 of NUREG-1505 or can be found in analysis of variance [ANOVA] output from commonly used statistical software (see Table G-2). The difference in concentration that is distinguishable from background may be expressed in terms of an appropriate multiple of ω. The equation below is based on equation 13-13 in NUREG-1505, where M is a multiplier selected using the DQO process and $n_o$ is equal to the number of measurements per reference area when the number of measurements in each reference area is the same.

$$LBGR = M\sqrt{\widehat{\omega}^2} = \mathrm{M}\sqrt{\frac{(S_b^2 - S_w^2)}{n_0}}$$

Equation 13-13 in NUREG-1505 can be used when the number of measurements in the reference areas are not the same. Using the DQO process, a value for M needs to be determined. Based on information contained in section 13.4 of NUREG-1505, a value of 3 ώ is a reasonable default. Using the ANOVA output in Table G-2, the LBGR = 3 x $\sqrt{\widehat{\omega}^2}$= 2.22. Note that the difference in means between reference areas 2 and 4 in Table G-1 is 1.82, which is consistent with the LBGR calculated based on 3 ώ.

**Table G-2 Analysis of Variance for Example Data**

| Source of Variation | Sum of Squares | Degrees of Freedom | Mean Square | F Statistic |
|---|---|---|---|---|
| **Between Groups** | 19.56 | 3 | 6.52 | 6.69 |
| **Within Groups** | 35.08 | 36 | 0.97 | |
| **Total** | 54.65 | 39 | | |

Section 9.5 of NUREG-1505 contains information on calculating the sample size for the Wilcoxon Rank Sum test under Scenario B, and is essentially the same method used in Scenario A. Also, the Wilcoxon Rank Sum test is performed similarly as it is used in Scenario A. The data for an example Wilcoxon Rank Sum test using Scenario B are shown in column A of Table G-3. In column B, the label "R" is inserted to denote a reference area measurement, and the label "S" to denote a survey unit measurement.

In Scenario B, the survey unit measurements are adjusted by subtracting the LBGR (e.g., 3 $\hat{\omega}$) from each survey unit measurement, whereas in Scenario A the reference area measurements are adjusted. Column C of Table G-3 contains the adjusted data obtained by subtracting the LBGR (142) from the survey unit measurements. The ranks of the adjusted data in Column C are listed in Column D. Next, the adjusted survey unit measurements are ranked, which range from 1 to 24, since there is a total of 12 + 12 = 24 measurements. The sum of all the ranks is N(N + 1)/2 = (24)(25)/2 = 300. Column E contains only the ranks belonging to the adjusted survey unit measurements. The sum of the ranks of the adjusted survey unit data, $W_s$, is 194.5.

When using the DQO process to determine an acceptable probability of a Type I error, it should be noted that the total probability of a Type I error in assessing the survey unit data is the sum of the probabilities of Type I errors for both the Wilcoxon Rank Sum and Quantile tests. Therefore, the acceptable probabilities of Type I errors in the Wilcoxon Rank Sum and Quantile tests is half the total acceptable probability. For this example, the acceptable total probability of a Type I error is set at 0.05, so $\alpha_{\mathrm{WRS}}$ is 0.025. For comparing the statistic $W_s$ with the critical value, Table A.4 of NUREG-1505 is used. However, for Scenario B, the meaning of m and n are the reverse of those in Scenario A. For Scenario B, m is the number of survey unit measurements and n is the number of reference area measurements in this table. From Table A.4 of NUREG-1505, for values of α for the Wilcoxon Rank Sum test = 0.025 and n = m = 12, the critical value is 184. Because the sum of the adjusted survey unit ranks, 194.5, is greater than the critical value, 184, the null hypothesis that the survey unit concentrations do not exceed the LBGR is rejected (i.e., the site is determined to be dirty). In this Scenario B example, the true survey unit residual radioactivity is judged to be in excess of 142 above background.

The Quantile test is only used in Scenario B and if the Wilcoxon Rank Sum test fails to reject the null hypothesis. Whereas the Wilcoxon Rank Sum test is best at detecting excess radioactivity that is uniformly distributed, the Quantile test is intended to recognize excess radioactivity in only a fraction of the survey unit (i.e., higher concentrations of radioactivity in a fraction of the survey unit). Although the null hypothesis was rejected and the Quantile test need not be performed, for this Scenario B example the Quantile test is presented for illustrative purposes

using the data in Table G-3. The data are adjusted and ranked in the same manner as in the Wilcoxon Rank Sum test, then the ranks are sorted, and the area associated with the rank is identified. The null hypothesis of the Quantile test is that there is no residual radioactivity above the LBGR in any part of the survey unit.

Table A.7 of NUREG-1505 provides for the Quantile test the critical value, k, of the largest r measurements for different values of n, the number of measurements from the survey unit, and m, the number of measurements from the reference area. Different sub-tables are provided in Table A.7 of NUREG-1505 for different α values. The same rankings in Column D of Table G-3 for the Wilcoxon Rank Sum test can be used for the Quantile test. If k or more of the r largest measurements in the combined ranked data set are from the survey unit, the null hypothesis is rejected.

Columns F and G of Table G-3 show the sorted ranks of the adjusted data, and the location associated with each rank (i.e., “R” for reference area and “S” for survey unit). From Table A.7b of NUREG-1505, the closest entry to n = m = 12 is for n = m = 10, with corresponding values of r = 7, k = 6 and α = 0.029. Thus, the null hypothesis is rejected if six of the seven largest adjusted measurements are from the survey unit. From Table G-3, we find that only five of the seven largest adjusted measurements are from the survey unit, so the null hypothesis is not rejected based on the Quantile test. The values of n and m that were used are close to, but not equal to, the actual values, so the α value will be different from that listed in the table. It is prudent to check a few other entries in Table A.7b of NUREG-1505 that are near the actual sample size. Additionally, Chapter 7 in NUREG-1505 provides equations to calculate exact and approximate values of α for the Quantile test as a function of n, m, k, and r.

It is recommended that an elevated measurement comparison is conducted, regardless of the outcome of the WRS and Quantile test. This consists of determining if any measurement in the remediated survey unit exceed a specified investigation level. If so, then additional investigation of the data is required to determine if there are elevated measurements that were not identified by the statistical tests.

**Table G-3 Scenario B WRS and Quantile Tests for Class 2 Interior Drywall Survey Unit**

| | A | B | C | D | E | F | G |
|---|---|---|---|---|---|---|---|
| **1** | **Data** | **Area** | **Adjusted Data** | **Ranks** | **Survey Unit Ranks** | **Sorted Ranks** | **Location Associated With Sorted Rank** |
| **2** | 47 | R | 47 | 18 | - | 1 | R |
| **3** | 28 | R | 28 | 1 | - | 2 | R |
| **4** | 36 | R | 36 | 6 | - | 3 | R |
| **5** | 37 | R | 37 | 7 | - | 4.5 | R |
| **6** | 39 | R | 39 | 9.5 | - | 4.5 | S |
| **7** | 45 | R | 45 | 13 | - | 6 | R |
| **8** | 43 | R | 43 | 11 | - | 7 | R |
| **9** | 34 | R | 34 | 3 | - | 8 | S |
| **10** | 32 | R | 32 | 2 | - | 9.5 | R |
| **11** | 35 | R | 35 | 4.5 | - | 9.5 | R |
| **12** | 39 | R | 39 | 9.5 | - | 11 | R |
| **13** | 51 | R | 51 | 21 | - | 13 | R |
| **14** | 209 | S | 67 | 24 | 24 | 13 | S |
| **15** | 197 | S | 55 | 23 | 23 | 13 | S |
| **16** | 188 | S | 46 | 16 | 16 | 16 | S |
| **17** | 191 | S | 49 | 19 | 19 | 16 | S |
| **18** | 193 | S | 51 | 21 | 21 | 16 | S |
| **19** | 187 | S | 45 | 13 | 13 | 18 | R |
| **20** | 188 | S | 46 | 16 | 16 | 19 | S |
| **21** | 180 | S | 38 | 8 | 8 | 21 | R |
| **22** | 193 | S | 51 | 21 | 21 | 21 | S |
| **23** | 188 | S | 46' | 16 | 16 | 21 | S |
| **24** | 187 | S | 45 | 13 | 13 | 23 | S |
| **25** | 177 | S | 35 | 4.5 | 4.5 | 24 | S |
| **26** | Sum= | | | 300 | 194.5 | | |

(Measurements from the reference area and the survey unit are denoted by R and S, respectively)

## G.7 Integration of Dose Modeling and Radiological Surveys

Pathway dose or risk modeling is oftentimes used to determine clean-up levels or DCGLs used as decision criteria in statistical tests discussed in Chapter 8 of MARSSIM and Appendix A. Because DCGLs are an integral part of the survey design, model integration with the survey design *is* an important topic discussed in various sections of this guidance.

Assuming the pathway dose or risk modeling is representative of actual survey unit conditions, the survey design should be compatible with modeling used to derive clean-up levels or DCGLs. Because the distribution, thickness, depth, and area of residual radioactivity are directly related to dose (or risk) in many commonly used decommissioning modeling codes, the FSS should be designed consistent with the modeling assumptions related to these key parameters to the extent practical. For example, if vertical heterogeneity is an issue and DCGLs are sensitive to the distribution of residual radioactivity within the soil column, DCGLs could be developed for different soil intervals (e.g., 0 to 5 cm and 5 to 15 cm). Depth discrete sampling may then be necessary for comparison against the DCGLs developed for each soil interval. For certain radionuclides and pathways (e.g., Cs-137 and external dose pathway), depth discrete sampling may also be important to ensure that higher concentration residual radioactivity located near the surface is not diluted with clean or cleaner radioactivity located deeper in the soil column, which could lead to an underestimate of risk[6]. Likewise, the thickness of contamination can also be important to risk. If residual radioactivity is located deeper in the soil column then assumed in the dose modeling, then the risk could be underestimated. Subsurface residual radioactivity should be adequately characterized and if present, appropriate methods should be developed to evaluate whether the site is clean (the MARSSIM methodology summarized in Appendix A was developed for surficial soils and building surfaces only; other methods may be necessary to make decisions regarding release for sites with subsurface or volumetric residual radioactivity).

Additionally, the dose and risk pathway modeling typically assume that contamination is relatively homogeneous across the model domain and most models only accept a single, average concentration for each radionuclide as input. If, in contrast, the survey unit is heterogeneous (e.g., spotty or elevated areas of radioactivity), the assumptions in the modeling may be violated, and the effectiveness of the statistical tests may be reduced. In these cases, the survey design team may need to consider other methods to mitigate or assess the impact of heterogeneous distributions such as more careful delineation of survey units or added emphasis and reliance on EMC tests. For Scenario B, the quantile test used in conjunction with the WRS test is also useful in evaluating whether a portion of the survey unit has unacceptable levels of contamination.

Finally, it is important to note that dose or risk modeling typically uses mean concentrations of radionuclides from survey unit measurements, while the Sign and WRS tests are tests on the median. While the median concentration is a good approximation for the mean concentration if the radionuclide distributions are symmetric (or not highly skewed), in some cases the mean concentration could exceed the median concentration. The average of the survey unit measurements (or the difference between the average survey unit and average reference area)

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6 While most dose modeling codes only accept a single concentration, lateral and vertical heterogeneity can be represented by running the model more than once. For example, to represent vertical heterogeneity, initial concentrations at the surface (e.g., 0 to 5 cm) can be run separate from deeper residual radioactivity (e.g., 5 to 15 cm).

should always be compared to the DCGL before a site is released (Scenario A) to help ensure that a site with residual radioactivity above the DCGL is not released.

## G.8 References


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# APPENDIX H

# CRITERIA FOR CONDUCTING SCREENING DOSE MODELING EVALUATIONS

## H.1 Introduction

This appendix consists of the technical guidance for the use of the screening criteria, applicable to Decommissioning Groups 1–3. Section I.8 of Appendix I contains details of the references cited below.

This section pertains to NRC staff's review of a licensee's demonstration of compliance with the dose criteria in 10 CFR Part 20, "Standards for Protection against Radiation," Subpart E, "Radiological Criteria for License Termination," using a screening approach dose analysis. The NRC staff should review the screening analysis using one or more of the currently available screening tools:

- a lookup table for common beta- and gamma-emitting radionuclides for building surface residual radioactivity (Volume 63 of the *Federal Register* (FR), page 64,132 (63 FR 64132); November 18, 1998)

- a lookup table for common radionuclides for soil surface residual radioactivity (64 FR 68395; December 7, 1999)

- screening levels derived using DandD Version 2.1, or the most current version, for the specific radionuclide(s) and using the DandD code's default parameters

Other tools for performing a screening analysis might become available in the future, or the NRC staff may modify current lookup tables or develop additional lookup tables (e.g., the NRC staff may develop lookup tables for the common alpha-emitters for building surfaces using modified parameter values in current versions of DandD[1]). In addition, the NRC staff may consider the use of other screening tools (e.g., other lookup tables or other conservative codes/models) after evaluating and comparing these screening tools with the current screening codes.

A licensee usually conducts a screening analysis for simple sites with building surface (i.e., nonvolumetric) and/or with surficial soil (approximately 15 centimeters (cm) (6 inches (in.)) residual radioactivity. The analysis usually employs simple and conservative models/codes and parameters, under generic exposure scenarios and default site conditions, to define the screening derived concentration guideline levels (DCGLs) equivalent to the dose criteria. Because of the conservative nature of the screening analysis approach, the screening DCGLs are typically more restrictive than the site-specific DCGLs. A screening analysis may save licensees time and effort by reducing the amount of site characterization, modeling analysis, and reviews needed, versus those needed when using a site-specific analysis approach.

To review a screening analysis, the NRC staff first needs to make a generic assessment and evaluation of a licensee's justification that the site is qualified for screening. In addition, the NRC staff should be familiar with the tools (e.g., models, codes, and calculations) and embedded assumptions used to derive the screening DCGLs. This section addresses the major

[1] One sensitive parameter identified by the NRC staff for the building occupancy exposure scenario is the resuspension factor. NUREG-1720, "Re-Evaluation of the Indoor Resuspension Factor for the Screening Analysis of the Building Occupancy Scenario for NRC's License Termination Rule," issued June 2002, documents the NRC staff's evaluation of the default resuspension factor used in DandD Version 1. If site conditions are consistent with the assumptions made in NUREG-1720, the recommended resuspension factor and parameter distribution in NUREG-1720 can be used with minimal justification in a site-specific analysis.

issues that the NRC staff may encounter in the generic screening analysis reviews and includes recommendations of approaches for addressing and resolving these issues.

### H.1.1 Issues in Performing Screening Analysis

The NRC staff may encounter issues with the screening analysis, including (1) the definition of screening and the transition from a screening to a site-specific analysis, (2) the qualification of the site for screening, in terms of site physical conditions and compatibility with the modeling code's assumptions and default parameters, and (3) the acceptable screening tools (e.g., code, lookup tables), approaches, and parameters that the staff can use to translate the dose into equivalent screening concentration levels. Each one of these issues is the subject of discussion in the following subsections:

### H.1.2 Screening Definition and Approaches for the Transition from Screening to Site-Specific Analysis

The NRC staff may encounter some inconsistencies about the definition of the term "screening" in dose analysis, which may cause confusion about the transition from a screening to a site-specific analysis. These inconsistencies become more apparent when dividing screening approaches into multiple levels (NCRP Report No. 123, "Screening Models for Releases of Radionuclides to Atmosphere, Surface Water, and Ground," dated January 22, 1996; NCRP Report No. 129, "Recommended Screening Limits for Contaminated Surface Soil and Review of Factors Relevant to Site-Specific Studies," dated January 29, 1999). In some cases, screening and site-specific terms are mixed, and the term "site-specific screening" is used (NUREG/CR-5512, Volume 1, "Residual Radioactive Contamination from Decommissioning: Technical Basis for Translating Contamination Levels to Annual Total Effective Dose Equivalent," issued October 1992). In certain cases, screening is categorized by the type of models used (e.g., simple and conservative models versus more advanced and complex models) and the extent of data and information needed to support the dose analysis.

Within the context of NUREG-1757, the NRC staff should consider the definition of screening as the process of developing DCGLs at a site using either (1) the NRC's lookup tables in 63 FR 64132 and 64 FR 68395, or (2) the latest version of the DandD code developed by the NRC to perform the generic screening analysis.

When licensees either (1) select other approaches or models for the dose analysis or (2) modify the DandD code default parameters, exposure scenarios, or pathways, the NRC staff considers licensees to be performing site-specific analyses. With regard to footnote "a" of Table H.1, the use of values of the fraction of removable surface contamination other than 0.1 or 1.0 (as described in the footnote) in the DandD code is considered a site-specific analysis and the staff should use Section 5.2 to review it.

While there is no requirement that licensees consider the use of screening criteria, they should recognize the advantages and disadvantages of selecting a screening approach for demonstrating compliance with the dose criteria. Section 2.6 of this volume discusses the merits of using screening versus using site-specific analysis.

### H.1.3 Qualification of the Site for Screening

The NRC staff should be aware that a screening analysis, for demonstrating compliance with the dose criteria in 10 CFR Part 20, Subpart E, may not be applicable for certain sites because of the status of contaminants (e.g., location and distribution of radionuclides), or because of site-specific physical conditions. Therefore, the staff should assess the source characteristics (e.g., spatial distribution of residual radioactivity) to ensure consistency with the source configuration assumptions in the DandD code. Further, the NRC may determine that there could be conditions at the specific site that cannot be handled by the simple screening model, because of the complex nature of the site or because of the simple conceptual model in the DandD screening code.

When using the screening approach for demonstrating compliance with the dose criteria in 10 CFR Part 20, Subpart E, licensees need to demonstrate that the particular site conditions (e.g., physical and source conditions) are compatible and consistent with the DandD model assumptions (NUREG/CR-5512, Volume 1). In addition, the default parameters, exposure scenarios, and pathways must also be used in the screening dose analysis. Therefore, reviewers should examine the site conceptual model, the generic source term characteristics, and other attributes of the site to ensure that it is qualified for screening.

The NRC staff should verify that the following site conditions exist for each of the residual radioactivity conditions:

- Building Surface Residual Radioactivity:
    - The residual radioactivity on building surfaces (e.g., walls, floors, ceilings) should be surficial and nonvolumetric (e.g., ≤10 millimeters (mm) (0.39 inches (in.)) of penetration).
    - Residual radioactivity on surfaces is mostly fixed (not loose), with the fraction of loose (removable) residual radioactivity no greater than 10 percent of the total surface activity. Note, for cases when the fraction of removable contamination is undetermined or higher than 0.1, licensees may assume, for screening purposes, that 100 percent of surface contamination is removable, and therefore the screening values should be decreased by a factor of 10 (see footnote “a” to Table H.1).
    - The screening criteria are not being applied to surfaces such as buried structures (e.g., drainage or sewer pipes) or equipment within the building without adequate justification; such structures, buried surfaces, and clearance of equipment should be treated on a case-by-case basis.
- Surface Soil Residual Radioactivity:
    - The initial residual radioactivity (after decommissioning) is contained in the top layer of the surface soil (e.g., approximately 15 cm (6.0 in.)).
    - Subsurface soil (e.g., approximately 15 cm (6.0 in.) or greater below the surface) in the unsaturated zone and the groundwater are initially free of residual radioactivity.
    - The vertical saturated hydraulic conductivity at the specific site is greater than the infiltration rate (e.g., there is no ponding or surface runoff).

Questions have also been raised about the appropriateness of using a screening analysis at sites with contaminated areas larger than the DandD Version 1 default cultivated area (e.g., 2,400 square meters ($m^2$) (25,800 square feet ($ft^2$)). Initially, the NRC staff evaluated the effect of a large contaminated area on the derived screening dose and determined that this effect is trivial for sites with the dominant dose arising from direct exposure or inhalation. As modeled by DandD with its default parameter set, this effect could be appreciable for sites with a significant dose contribution associated with the ingestion pathway (specifically ingestion associated with the drinking water and irrigation pathways). The staff determined that, for sites with contaminated areas of 6,000–7,200 $m^2$ (64,600–77,500 $ft^2$), the dose may be underestimated under worst-case conditions by a factor of 2 to 3. However, further staff analysis showed that, because of the conservative assumptions of the DandD code, it is more likely that the derived dose (based on the use of other codes or the use of a site-specific analysis) would be far less than the derived dose using these default conditions. Therefore, for sites with areas larger than 7,200 $m^2$ (77,500 $ft^2$), the change in actual risk due to this effect is not appreciable. In summary, assuming that the site is qualified for screening based on the above-listed criteria, the NRC would accept the screening approach for sites with areas larger than the default cultivated area (i.e., 2,400 $m^2$ (25,800 $ft^2$)).

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It should be noted that the NRC staff should also evaluate complex site conditions that may disqualify the site for screening. Examples of such complex site conditions may include highly fractured formation, karst conditions, extensive surface water contamination, and highly nonhomogeneous distribution of residual radioactivity. Therefore, reviewers should ensure that the site meets the definition of a “simple site” to qualify for screening (see Section 1.2 of this document for additional details).

### H.1.4 Acceptable Screening Tools

In the past, it may not have been clear what screening tools the NRC has determined to be acceptable. Some may believe that using simple, common codes (other than DandD), with their deterministic default parameters, may be acceptable to derive the desired screening derived concentration guideline levels (DCGLs). Others may believe that the use of any lookup tables published by certain scientific committees or authorities may be used to convert concentration levels directly into doses for purposes of complying with Subpart E. Questions have also been raised on the use of the DandD code for screening, particularly whether modifying input default parameters is acceptable for screening.

The NRC staff should accept, for screening analyses, the following currently available screening tools:

- a lookup table (Table H.1) for common beta- and gamma-emitting radionuclides for building-surface residual radioactivity (63 FR 64132; November 18, 1998)
- a lookup table (Table H.2) for common radionuclides for soil surface residual radioactivity (64 FR 68395; December 7, 1999)
- screening levels derived using the latest version of DandD for the specific radionuclide and using code default parameters and parameter ranges

The screening values in Tables H.1 and H.2 are intended for single radionuclides. For radionuclides in mixtures, the “sum of fractions” rule should be used (see Section 2.7 of this volume). Table H.1 values that provide screening values for beta and gamma emitters

associated with building surfaces were derived using DandD Version 1 and its deterministic, default parameter set, before the development of probabilistic DandD Version 2. Table H.2 values were derived using a version of DandD that was similar to and the predecessor of DandD Version 2. The screening values in Table H.2 are based on the selection of the 90th percentile of the output dose distribution for each specific radionuclide or radionuclide with the specific decay chain. Behavior parameters were set at the mean of the distribution of the assumed critical group. The metabolic parameters were set either at the Reference Man or at the mean of the distribution for an average human.

For a radionuclide with its progeny present at equilibrium, the "+C" values of Table H.2 should be interpreted carefully. As described in footnote "c" to Table H.2, these "+C" values are concentrations of the parent radionuclide only but account for dose contributions from the complete chain of progeny in equilibrium with the parent radionuclide. For example, U-238+C lists the soil screening value as 18.5 Bq/kg (0.5 pCi/g). This means that it also assumed the presence of 18.5 Bq/kg (0.5 pCi/g) of U-234, 18.5 Bq/kg (0.5 pCi/g) of Th-230, and so forth.

The current NRC staff position is to limit screening to its lookup tables or the execution of the latest version of DandD code with the default parameters and distributions. As indicated above, the NRC staff may develop additional lookup tables or modify the screening tables based on refining certain sensitive parameters in the future. For example, NUREG-1720 documents the NRC staff's evaluation of the default resuspension factor used in DandD Version 1.0. NUREG-1720 provides specific recommendations related to the use of the resuspension factor in screening analyses for the building occupancy exposure scenario. If site conditions are consistent with assumptions in NUREG-1720 (with respect to activities and scenarios, ventilation conditions, and low removable fractions at the time of decommissioning), the NRC staff has determined that it is acceptable to use the NUREG-1720-recommended resuspension factor value or parameter distribution with minimal justification. However, until such time as the NUREG-1720-recommended parameter value or distribution is used in developing an updated screening table for the building occupancy exposure scenario, the NRC staff considers use of the NUREG-1720 values a site-specific analysis (because the underlying assumptions in NUREG-1720 must be verified by the licensee before use of the recommended resuspension factor values). The NUREG-1720-recommended resuspension factor is approximately 10 times less than the default value used in DandD Version 1 and would, thus, lead to significantly higher (or less restrictive) screening values for radionuclides dominated by the inhalation pathway (e.g., alpha-emitting radionuclides), had such screening values been published in 63 FR 64132[2].

ORISE (2017) provides screening values for sites potentially contaminated with discrete sources of Ra-226 developed using the DandD code and three exposure scenarios including (i) industrial building occupancy, (ii) residential building occupancy, for residual radioactivity on building surfaces; and (iii) resident farmer for residual radioactivity associated with soils. Site-specific dose modeling is required for sites with groundwater contamination or with site conditions that are otherwise inconsistent with the dose modeling assumptions. The screening values are stated to be inappropriate for use for other types of Ra-226 contaminated decommissioning sites.

[2] The NRC staff only published beta- and gamma-emitting radionuclide screening values in 63 FR 64132, while indicating it was continuing its assessment of screening approaches for sites with alpha-emitting radionuclides.

**Table H.1 Acceptable License Termination Screening Values of Common Radionuclides for Building-Surface Contamination**

| Radionuclide | Symbol | Acceptable Screening Levels[a] for Unrestricted Release (dpm/100 $cm^2$)[b] |
|---|---|---|
| Hydrogen-3 (Tritium) | $^{3}H$ | 120000000 |
| Carbon-14 | $^{14}C$ | 3700000 |
| Sodium-22 | $^{22}Na$ | 9500 |
| Sulfur-35 | $^{35}S$ | 13000000 |
| Chlorine-36 | $^{36}Cl$ | 500000 |
| Manganese-54 | $^{54}Mn$ | 32000 |
| Iron-55 | $^{55}Fe$ | 4500000 |
| Cobalt-60 | $^{60}Co$ | 7100 |
| Nickel-63 | $^{63}Ni$ | 1800000 |
| Strontium-90 | $^{90}Sr$ | 8700 |
| Technetium-99 | $^{99}Tc$ | 1300000 |
| Iodine-129 | $^{129}I$ | 35000 |
| Cesium-137 | $^{137}Cs$ | 28000 |
| Iridium-192 | $^{192}Ir$ | 74000 |

Notes:

a. Screening levels are based on the assumption that the fraction of removable surface contamination is equal to 0.1. For cases when the fraction of removable contamination is undetermined or higher than 0.1, users may assume for screening purposes that 100 % of surface contamination is removable, and therefore the screening levels should be decreased by a factor of 10. Users may calculate site-specific levels using available data on the fraction of removable contamination in DandD.[3]

b. Units are dpm/100 $cm^2$. One dpm is equivalent to 0.0167Bq. To convert to units of $Bq/m^2$, multiply each value by 1.67. The screening values represent surface concentrations of individual radionuclides deemed to be in compliance with the 0.25 mSv/y (25 mrem/y) unrestricted release dose limit in 10 CFR 20.1402. For radionuclides in a mixture, the "sum of fractions" rule applies (see note 4, of footnote 3 to 10 CFR Part 20, Appendix B for a description of sum of fractions. An unofficial copy of the note is available on the NRC website at https://www.nrc.gov/reading-rm/doc-collections/cfr/part020/appb/footnotes.html).

c. NUREG/CR-5512, Vol. 3, Table 5.19 (see values in column labeled $P_{crit}$ = 0.90), contains screening values for additional radionuclides not found in Table H.1. The Table 5.19 screening values are also acceptable for use provided the underlying assumptions of the screening approach are met (65 FR 37186).

3 The original footnote to this table, published in 1998 (63 FR 64132), referenced use of DandD Version 1, which was the version of the code used to develop the screening values in this table. However, after publication of the screening values in the FR in 1998, probabilistic DandD Version 2 was developed and was used to create the screening values reported in Table H.2. Therefore, licensees should use more current versions of DandD (Version 2 and later) to develop screening values.

**Table H.2 Screening Values[a] (pCi/g) of Common Radionuclides for Soil Surface Contamination Levels**

| Radionuclide | Symbol | Surface Soil Screening Values[b] |
|---|---|---|
| Hydrogen-3 | $^{3}H$ | 110 |
| Carbon-14 | $^{14}C$ | 12 |
| Sodium-22 | $^{22}Na$ | 4.3 |
| Sulfur-35 | $^{35}S$ | 270 |
| Chlorine-36 | $^{36}Cl$ | 0.36 |
| Calcium-45 | $^{45}Ca$ | 57 |
| Scandium-46 | $^{46}Sc$ | 15 |
| Manganese-54 | $^{54}Mn$ | 15 |
| Iron-55 | $^{55}Fe$ | 10000 |
| Cobalt-57 | $^{57}Co$ | 150 |
| Cobalt-60 | $^{60}Co$ | 3.8 |
| Nickel-59 | $^{59}Ni$ | 5500 |
| Nickel-63 | $^{63}Ni$ | 2100 |
| Strontium-90 | $^{90}Sr$ | 1.7 |
| Niobium-94 | $^{94}Nb$ | 5.8 |
| Technetium-99 | $^{99}Tc$ | 19 |
| Iodine-129 | $^{129}I$ | 0.5 |
| Cesium-134 | $^{134}Cs$ | 5.7 |
| Cesium-137 | $^{137}Cs$ | 11 |
| Europium-152 | $^{152}Eu$ | 8.7 |
| Europium-154 | $^{154}Eu$ | 8 |
| Iridium-192 | $^{192}Ir$ | 41 |
| Lead-210 | $^{210}Pb$ | 0.9 |
| Radium-226 | $^{226}Ra$ | 0.7 |
| Radium-226+C[c] | $^{226}Ra$+C | 0.6 |
| Actinium-227 | $^{227}Ac$ | 0.5 |
| Actinium-227+C | $^{227}Ac$+C | 0.5 |
| Thorium-228 | $^{228}Th$ | 4.7 |

**Table H.2 Screening Values (pCi/g) of Common Radionuclides for Soil Surface Contamination Levels (cont.)**

| Radionuclide | Symbol | Surface Soil Screening Values[a,b] |
|---|---|---|
| Thorium-228+C[c] | $^{228}$Th+C | 4.7 |
| Thorium-230 | $^{230}$Th | 1.8 |
| Thorium-230+C | $^{230}$Th+C | 0.6 |
| Thorium-232 | $^{232}$Th | 1.1 |
| Thorium-232+C | $^{232}$Th+C | 1.1 |
| Protactinium-231 | $^{231}$Pa | 0.3 |
| Protactinium-231+C | $^{231}$Pa+C | 0.3 |
| Uranium-234 | $^{234}$U | 13 |
| Uranium-235 | $^{235}$U | 8 |
| Uranium-235+C | $^{235}$U+C | 0.29 |
| Uranium-238 | $^{238}$U | 14 |
| Uranium-238+C | $^{238}$U+C | 0.5 |
| Plutonium-238 | $^{238}$Pu | 2.5 |
| Plutonium-239 | $^{239}$Pu | 2.3 |
| Plutonium-241 | $^{241}$Pu | 72 |
| Americium-241 | $^{241}$Am | 2.1 |
| Curium-242 | $^{242}$Cm | 160 |
| Curium-243 | $^{243}$Cm | 3.2 |

Notes:

a. These values represent surficial surface soil concentrations of individual radionuclides that would be deemed in compliance with the 0.25 mSv/y (25 mrem/y) unrestricted release dose limit in 10 CFR 20.1402. For radionuclides in a mixture, the "sum of fractions" rule applies; see Section 2.7 of this volume.

b. Screening values are in units of (pCi/g) equivalent to 0.25 mSv/y (25 mrem/y). To convert from pCi/g to units of Bq/kg, divide each value by 0.027. These values were derived using DandD screening methodology (NUREG/CR–5512, Volume 3 (NRC 1999)). They were derived based on selection of the 90th percentile of the output dose distribution *for each specific radionuclide* (or radionuclide with the specific decay chain). Behavioral parameters were set at the mean of the distribution of the assumed critical group. The metabolic parameters were set at Reference Man or at the mean of the distribution for an average human.

c. "Plus Chain (+C)" indicates a value for a radionuclide with its decay progeny present in equilibrium. The values are concentrations of the parent radionuclide but account for contributions from the complete chain of progeny in equilibrium with the parent radionuclide (NUREG/CR–5512, Volumes 1, 2, and 3).

d. NUREG/CR-5512, Vol. 3, Table 6.91 (see values in column labeled $P_{crit}$ = 0.10), contains screening values for additional radionuclides not found in Table H.2. The Table 6.91 screening values are also acceptable for use provided the underlying assumptions of the screening approach are met (65 FR 37186).

## H.2 References

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# APPENDIX I

# TECHNICAL BASIS FOR SITE-SPECIFIC DOSE MODELING EVALUATIONS

# I.1 Introduction

This appendix consists of the technical guidance for the use of site-specific dose modeling, applicable to Decommissioning Groups 4–7.

## I.1.1 Background

On July 21, 1997, the NRC published a final rule on "Radiological Criteria for License Termination," in the *Federal Register* (62 FR 39058), which was incorporated as Subpart E, "Radiological Criteria for License Termination," to 10 CFR Part 20, Standards for Protection against Radiation." In 1998, the NRC staff developed a draft regulatory guide (DG), "Demonstrating Compliance with the Radiological Criteria for License Termination" (DG–4006) (NRC, 1998e), and a draft document, NUREG-1549, "Decision Methods for Dose Assessment to Comply with Radiological Criteria for License Termination" (NRC, 1998d), in support of the final rule. In addition, the staff developed a screening code "DandD" for demonstrating compliance with the dose criteria in 10 CFR Part 20, Subpart E.

On July 8, 1998, the Commission approved publication of the draft guidance, DG-4006, the draft NUREG-1549, and the DandD screening code for interim use for a 2-year period (i.e., from July 8, 1998, through July 7, 2000), in "Staff Requirements—SECY–98–051—'Guidance in Support of Final Rule on Radiological Criteria for License Termination,'" dated July 8, 1998 (NRC, 1998c). In addition, the Commission directed the staff to develop a SRP for decommissioning and provide the Commission with a timeline for developing it, maintain a dialogue with the public during the interim period, address areas of excessive conservatism, particularly in the DandD screening code, develop a more user-friendly format for the guidance, and use a probabilistic approach to calculate the TEDE to the average member of the critical group (NRC, 1998c).

The NRC staff published NUREG-1727, "NMSS Decommissioning Standard Review Plan," in September 2000. Chapter 5 of the SRP (which is incorporated into Chapter 5 of this volume) addresses the NRC staff review of the licensee's dose modeling to demonstrate compliance with the criteria in 10 CFR Part 20, Subpart E. The NRC staff developed Appendix C of the SRP (Appendix I of this volume) as a technical information support document for performing staff evaluations of the licensee's dose modeling. It presents detailed technical approaches, methodologies, criteria, and guidance to the staff reviewing dose modeling for demonstrating compliance with the dose criteria in 10 CFR Part 20, Subpart E. To develop Appendix C of the SRP, the NRC used an iterative process with the public, which included licensees, Federal agencies, States, and other interested individuals. To support this process, the staff conducted seven public workshops and gave several presentations to national and international professional groups, stakeholders, the Interagency Steering Committee on Radiation Standards, and the Conference of Radiation Control Program Directors, as well as to the NRC's Advisory Committee on Nuclear Waste. In addition, the NRC posted the draft Appendix C (formerly the Technical Basis Document) on the NRC's website and invited interested individuals to comment.

Since the publication of the license termination rule (LTR), the NRC staff has tested the DandD code for complex sites and addressed the issue of excessive conservatism in the DandD code. In addition, the NRC developed a new probabilistic DandD code (i.e., DandD Version 2.1) to reduce the excessively conservative approach in the initial version. Further, the staff developed the RESRAD-ONSITE and RESRAD–BUILD probabilistic codes for site-specific analysis, which also responds to the Commission's direction to use a probabilistic approach to calculate the

TEDE to the average member of the critical group. Later, RESRAD-OFFSITE was developed and also included probabilistic capabilities.

**Licensees using probabilistic dose modeling should use the "peak of the mean" dose (see Section I.7.3.2 from Appendix I of this volume) for demonstrating compliance with 10 CFR Part 20, Subpart E. Similar to all regulatory guidance, this NUREG contains one approach acceptable to the NRC staff for determining compliance with the regulations using probabilistic analyses. Use of "mean of the peaks" is also acceptable for demonstrating compliance. If the "mean of the peaks" dose is significantly higher than the "peak of the mean" dose, "risk dilution" may be an issue in the probabilistic model. Consult Appendix Q for more information on the potential for and impacts of "risk dilution." To use any probabilistic approach to calculate DCGLs, the licensee should discuss its approach with the NRC staff.**

### I.1.2 Brief Description and Scope

This section is divided into the following different topic areas, as summarized below.

- Section I.2 presents NRC approaches for reviewing the conceptual representation of the distribution and release of residual radioactivity from soils and building materials. This section describes areas of review for conceptualization and representation of the source and source term in dose modeling used to demonstrate compliance with radiological criteria for license termination

- Section I.3 focuses on areas of review and criteria for modifying the two generic exposure scenarios used in screening-level analyses: modifications to (1) the "resident farmer" and (2) the "building occupancy" exposure scenarios. Section I.3, and Appendices L and M, discuss the type of information a licensee should provide to support the modification of default (screening) exposure scenarios. Section I.3 also presents approaches for establishing site-specific exposure scenarios and pathways reflecting the activities and behaviors of identified critical group(s), based on current and reasonably foreseeable future land use, site restrictions, and other physical conditions associated with a decommissioning site.

- Section I.4 provides approaches for developing conceptual and mathematical models in dose modeling. This section presents approaches for collecting and using characterization data to develop conceptual and mathematical models of the site and issues associated with model simplification and abstraction. This section also includes information on the underlying conceptual models in the DandD and RESRAD codes and associated limitations.

- Section I.5 presents approaches and criteria for NRC staff acceptance of computer codes/models. This section discusses review aspects pertaining to specifications, testing, verification, documentation, and QA/QC of the licensee's codes and models. This section also addresses reviews applicable to embedded numerical models for the source term, fate and transport, and biosphere (exposure) modeling.

- Section I.6 describes approaches for the selection and modification of input parameters used in dose modeling.

- Section I.7 provides information on evaluating uncertainty and identifying parameters most important to dose in analyses used to demonstrate compliance with LTR criteria.

- Section I.8 compiles the references used throughout the appendix.

## I.2 Source and Source Term Abstraction[1]

### I.2.1 Introduction

Source abstraction is the process of developing a conceptual representation of the residual radioactivity present at a site, focusing on the geometry and distribution of contamination in the environment. Typically, the radiological conditions at a site undergoing decommissioning are relatively complex. Source abstraction is necessary to allow the detailed radiological characterization of the site to be incorporated into the mathematical and computer models that are used to demonstrate compliance with the dose-based criteria in 10 CFR Part 20, Subpart E. The abstraction process involves generalizing the radiological characteristics across the site to produce a simplified representation, which should facilitate the modeling of radiological impacts. This guidance makes a distinction between parameters that characterize the source, including parameters related to the concentration of radionuclides, thickness, and area of residual radioactivity; and those parameters that characterize the release of radioactivity in the environment (i.e., that help define the source term). The source term considers parameters that define the source concentrations and configuration of residual radioactivity, as well as the release mechanism (e.g., solubility–controlled, desorption, or diffusion-limited release). Although source and source term abstraction are a necessary part of the dose modeling process, the licensee should take care to ensure that the conceptual representation of the source and source term developed in the abstraction process are not oversimplified in a manner that results in the underestimation of potential radiological impacts.

As discussed in Chapter 5 of this volume, source configuration and source term abstraction serve as the starting point for the dose modeling process. The conceptual abstraction of the source and source term is combined with (1) the physical characteristics of the site, and (2) characteristics of the average member of the critical group to develop a conceptual model for the site being studied. Thus, the conceptual model also provides a representation of the natural environment through which radioactivity can be transported, as well as applicable exposure scenarios and pathways to members of the public who may be exposed to the radioactivity. The conceptual model can be used to determine acceptability of use of computer codes that contain their own built-in conceptual models with respect to how residual radioactivity is transported in the environment and how humans can be exposed to radioactivity.

[1] Source abstraction or configuration takes into consideration the geometry of the source (e.g., areal extent and thickness), as well as the distribution of residual radioactivity within the extent of contamination (e.g., homogenous versus nonhomogeneous). On the other hand, the “source term” characterizes the release rate of radionuclides from the source zone. The source term is a function of the inventory, physical and chemical characteristics of the contaminated materials, and surrounding environment, as well as the release mechanism (e.g., solubility-controlled, desorption, or diffusion-limited release). RESRAD-ONSITE, RESRAD-BUILD, and DandD have built-in release mechanisms and models, while RESRAD-OFFSITE offers several additional options to define the “source term.” Note that the working definition of source term in this volume is slightly different than the definition of source term found in the NRC glossary. The NRC glossary definition of source term, which is specific to accidents involving radioactive materials, indicates the following: “types and amounts of radioactive or hazardous material released to the environment following an accident”.

Volume 1 of NUREG-1757 and Chapter 4 of this volume discuss the information the licensee is expected to provide about the existing radiological characterization of the site. The licensee should describe the types, levels, and extent of residual radioactivity in contaminated materials at the site. This should include residual radioactivity in all media (including buildings, systems, and equipment that will remain after license termination; surface and subsurface soil; and surface water and groundwater). The source configuration and source term abstraction should be based on the characterization of the radiological status (e.g., historical site assessment; records of leakage or disposal). The licensee should explicitly relate the information provided in the discussion of radiological status of the site with its assumptions on the source configuration in dose modeling. The reviewer should be able to clearly interpret the relationship.

Generally, in the source abstraction process, the licensee may focus on several specific elements, which include the following:

- The licensee should identify the radionuclides of concern, taken directly from the description of the site's radiological status. The licensee should identify the radionuclides based on the pre-remediation radiological status. It should include all radionuclides potentially present at the site, so that their presence or absence may be verified during the FSS, except as noted in Chapter 4, "Facility Radiation Surveys," and Section 3.3, "Insignificant Radionuclides and Exposure Pathways," of this volume.

- The licensee should describe the physical/chemical form(s) of the contaminated media *anticipated at the time of FSS and site release*. The licensee should indicate whether the residual radioactivity will be limited to building surfaces or surface soil, or both, or whether the residual radioactivity will involve other media, such as subsurface soil, debris or waste materials (e.g., sludge, slag, tailings), or ground and surface water. Information on the physical/chemical form(s) of the contaminated media will also help determine whether source term assumptions are appropriate.

- The licensee should delineate the spatial extent of the residual radioactivity *anticipated at the time of FSS and site release*. The delineation of the spatial extent should include descriptions of (1) the areal extent of radionuclides throughout the site and (2) the vertical extent of soil residual radioactivity of radionuclides below the ground surface. The delineation of spatial extent and depth should establish the source areas and volumes. Source areas and volumes may differ for individual radionuclides.

- The licensee should define the distribution of each radionuclide throughout the delineated source areas and volumes *anticipated at the time of FSS and site release*. The distribution of a radionuclide through the source should be defined in terms of representative volumetric or areal concentrations. In addition, for volumetrically contaminated soil, the licensee may provide an estimate of total activity of each radionuclide.

- The licensee should define sources in surface water or groundwater, if any, based on environmental monitoring and sampling of aquifers and surface water bodies. A site with groundwater or surface water contamination may be categorized as "complex" and may require more advanced dose modeling analysis (see Appendix F for additional information on surface water and groundwater characterization).

In the source abstraction process, the licensee should identify the radionuclides of concern and have sufficient information to determine if residual radioactivity is surficial and relatively

homogeneous. Depending on the dose modeling approach, the licensee may or may not need to address the other elements, as discussed in more detail later in this section.

### I.2.2 Issues Associated with Source Term Abstraction

The level of effort that a licensee expends to develop a conceptualization of the source and source term should be commensurate with the complexity of the site and the licensee's approach to demonstrating compliance with the release criterion (i.e., screening versus site-specific). Also, the focus should be on the source and source term characteristics anticipated to exist at the site at the time of FSS and release, after any planned remediation.

If a licensee plans to use the screening DCGLs published by the NRC in the *Federal Register* (see Appendix H), a licensee should only have to identify the radionuclides present at the site and demonstrate that the conditions at the site meet the prerequisites for using the screening values (i.e., residual radioactivity is limited to building surfaces or approximately the uppermost 15 centimeters (cm) (6 inches (in.)) of surface soil and show there is no contamination of groundwater or surface water[2]), as discussed further in Section I.2.3 of this appendix.

If a licensee anticipates that residual radioactivity will be limited to building surfaces or surface soils at the time of the FSS but considers the published DCGLs overly restrictive, it may develop site-specific DCGLs. In this case, the licensee would most likely have to delineate the anticipated areal extent and depth of residual radioactivity. However, the licensee would not have to discuss the anticipated spatial variability of radionuclide concentrations within the anticipated area of residual radioactivity in developing the DCGL, because variability is considered in the survey design and in EMC tests, as discussed in more detail below.[3]

The licensee should provide a site-specific dose assessment if the residual radioactivity is not limited to building surfaces or surface soil. In this case, the licensee would have to delineate the spatial extent (laterally and vertically) of the residual radioactivity and discuss the spatial variability of the physical, chemical, and radiological characteristics of the contaminated media.

Ideally, the source characteristics at a site would be relatively uniform, justifying simplified source abstraction. However, this is often not the case. Issues may arise when the residual radioactivity projected at a site at the time of release is inconsistent with the ideal case. These issues may include the following:

(1) Spatial extent

- o limited areal extent of residual radioactivity
- o irregular areal shape

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2 If surface water or groundwater are contaminated, it may still be possible to use screening values if the dose contributions from the surface water or groundwater contamination are separately considered.

3 It is important to note that if insignificant radionuclides or modified DCGLs (e.g., surrogate DCGLs) are developed to account for multiple radionuclides, it would be important for the licensee to discuss spatial variability in radionuclide ratios used to develop surrogate relationships to ensure that the modified DCGLs do not lead to an underestimate of risk. However, this information is unnecessary for the development of site-specific DCGLs. See Appendix A, and Section 4.3.2 of MARSSIM, Revision 1, for more information on the use of surrogate radionuclides.

- varying depth of residual radioactivity in soil

(2) Spatial variability

- nonuniform distribution of radioactivity throughout a site
- limited areas of relatively elevated radionuclide concentrations
- multiple noncontiguous areas of residual radioactivity
- nonuniform physical and chemical characteristics

The sections below discuss approaches to addressing these technical issues.

### I.2.3 Approach to Source Abstraction

Source abstraction data needs will depend on the approach used to demonstrate compliance with radiological criteria for license termination presented in the licensee's DP. Generally, the licensee will use one of the two following approaches to dose modeling:

(1) Develop DCGLs for each radionuclide that would lead to a dose at the release criterion, and then demonstrate through the FSS that median residual radioactivity concentrations at the site are equal to or below the DCGLs with a certain specified level of confidence.

(2) Assess dose associated with the actual distribution of residual radioactivity at the site to determine whether the residual radioactivity will result in a dose equal to or below the dose-based release criterion.

In the first approach, the licensee intends to demonstrate at the time of the FSS that residual radionuclide concentrations across the site are below a prespecified concentration limit with some prespecified degree of confidence. The design of the FSS would be based on the proposed DCGLs, in accordance with NUREG-1575, "Multi-Agency Radiological Survey and Site Investigation Manual (MARSSIM)" (NUREG, 2000a). The MARSSIM process considers variability in determining the number of samples needed to demonstrate compliance with the radiological criteria for license termination. $DCGL_{EMC}$s are used to account for the dose contributions of smaller elevated areas of residual radioactivity. Knowledge about the characteristics (e.g., area and thickness) of elevated areas may also assist with the development of $DCGL_{EMC}$s.

In the second approach, the licensee intends to assess potential radiation doses that may result from specified levels of radioactive material. The contaminated material may not be limited to building surfaces or surface soils but may include contaminated subsurface soil, debris, and waste. The licensee's dose modeling should demonstrate that the residual radioactivity should not result in radiation doses in excess of applicable regulatory limits. Most likely, this modeling approach would require that the licensee incorporate information on both the spatial extent and spatial variability of radioactivity in the source abstraction.

Table I.1 summarizes source abstraction information needs, depending on the licensee's dose modeling approach and whether the licensee is providing screening or site-specific analyses. This table can serve as an index for the reviewer of the licensee's source abstraction.

**Table I.1 Summary of Source Abstraction Information Needs for Two Types of Dose-Modeling Approaches (Screening versus Site-Specific)**

| Approach to Demonstrating Compliance | Screening | Site-Specific |
|---|---|---|
| **DCGLs** | No source term abstraction is necessary beyond radionuclide identification.<br>(assume unit radionuclide concentrations to calculate DCGLs (i.e., to calculate what concentration leads to a dose at the limit)) | Delineate proposed lateral and vertical extent of residual contamination. Smaller areas of elevated activity can be considered through $DCGL_{EMC}$s.<br>(assume unit radionuclide concentrations to calculate DCGLs (i.e., to calculate what concentration leads to a dose at the limit)) |
| **Dose Modeling** | Use actual concentrations with DandD Version 2 (or later version) and ensure that residual radioactivity is surficial and spatial variability is minimal. | Site–specific source abstraction incorporates spatial extent and variability. |

### *I.2.3.1 Dose Modeling Approach One: Develop DCGLs*

The MARSSIM approach, as documented in NUREG-1575 (NRC, 2000a) and discussed in Chapter 4 of this volume, requires that a licensee establish a set of DCGLs before conducting an FSS. In fact, the design of the FSS should be based on the identified DCGLs. DCGL is defined in MARSSIM as:

> "*…a derived, radionuclide-specific activity concentration within a survey unit corresponding to the release criterion….DCGLs are derived from activity/dose relationships through various exposure pathway scenarios.*"

The $DCGL_W$ is the concentration of a radionuclide that, if distributed uniformly across a survey unit, would result in an estimated dose equal to the applicable dose limit. The $DCGL_{EMC}$ is the concentration of a radionuclide that, if distributed uniformly across a smaller limited area within a survey unit, would result in an estimated dose equal to the applicable dose limit. $DCGL_{EMC}$ is specific to the size of the area for which it is derived.

Two approaches are possible for developing DCGLs: screening and site-specific analysis.

*SCREENING DCGLs*

The NRC has published radionuclide-specific screening DCGLs in the *Federal Register* for residual building-surface radioactivity and residual surface-soil radioactivity (see Appendix H, Table H-1 and H-2; or Table 5.19 and 6.91 in NUREG/CR-5512, Volume 3 for additional radionuclides). The DCGLs in the *Federal Register* are intended to be concentrations that, if distributed uniformly across a building or soil surface, would individually result in a dose equal to

the dose criterion. The licensee may adopt these screening DCGLs without additional dose modeling, if the site is suitable for screening analysis. Alternatively, the licensee may use the DandD computer code to develop screening DCGLs. The licensee would use the code to determine the dose attributable to a unit concentration of a radionuclide and scale the result to determine the $DCGL_W$ for the radionuclide. Either of these methods for identifying screening DCGLs requires the licensee (1) to identify the radionuclides of concern for the site and (2) to demonstrate that the source and model screening assumptions are satisfied. Thus, this approach requires essentially no source abstraction.

Typically, before designing an FSS, the licensee identifies a $DCGL_{EMC}$ for each radionuclide for smaller limited areas of radioactivity (e.g., for areas between sampling locations). However, the underlying assumption for use of screening analyses is that the residual radioactivity is homogeneous. By default, DandD calculates dose based on the assumption that the size of the contaminated area is "unlimited." Although DandD provides an option to allow the user to enter a "limited area" of contamination, due to the simplistic manner in which DandD corrects the dose for smaller, limited areas of contamination, DandD is not ideally suited for calculating $DCGL_{EMC}$s. Therefore, it is recommended that licensees use other codes or approaches to develop $DCGL_{EMC}$ values, if there is a need for allowing higher concentrations above the $DCGL_W$ in smaller areas between sampling locations. These would be considered "site-specific" analyses in that they would not be using the DandD code with the default screening values. Section I.3.3.3.5 of this appendix contains additional information on this topic. If the licensee can show that residual radioactivity is relatively homogeneous and concentrations of radionuclides averaged over relatively small exposure areas (areas that are consistent with the exposure pathway assumptions in DandD) are less than the screening values or DandD-derived DCGLs, then development of $DCGL_{EMC}$ values should not be needed. Consult the DandD online help for more information on calculating the average concentrations for use in the DandD code.

*SITE-SPECIFIC DCGLs*

The licensee may choose to identify site-specific DCGLs if (1) the site conditions are not consistent with screening criteria or (2) the licensee believes the screening DCGLs are unnecessarily restrictive. As defined in MARSSIM, Rev. 1 (2000a) the licensee may derive site-specific DCGLs from activity or dose relationships through various exposure pathway scenarios. "Site-specific" in this context may refer to the selection of conceptual models/computer models, physical (site) input parameter values, or behavioral or metabolic input parameter values. These aspects of site-specific analyses are discussed in other sections of this document. "Site-specific" may also refer to the source or source term abstraction.

From the MARSSIM perspective, identifying a site-specific $DCGL_W$ still begins with assuming a uniform radionuclide concentration across some source area (building surface) or volume of surface soil[4]. The site-specific $DCGL_W$ for a particular radionuclide may be identified by evaluating the dose resulting from a unit concentration and then scaling the result to the dose limit. Spatial variability of the radionuclide concentration within the area or volume is not evaluated in calculating the DCGLs but is taken into account in the statistical analysis of the data collected during the FSS. When developing the site-specific DCGLs, the licensee should, however, take the spatial extent into account, including the horizontal and vertical extent of

[4] MARSSIM only considers surface soils; residual radioactivity in subsurface soils is not addressed by MARSSIM. Surface soil is defined in MARSSIM as "the top layer of soil onsite that is available for direct exposure, growing plants, resuspension of particles for inhalation, and mixing from human disturbances."

contamination. While some examples in MARSSIM use the top 15 cm of soil as surface soil, for the purpose of compliance measurements, licensees should consider the actual depth of residual radioactivity which should be consistent with the dose model used to generate site DCGLs. Soil sampling should be representative of the chosen surface soil depth, and surface scanning methods should be capable of detecting contamination within the entire depth of surface soil. Accordingly, scan MDCs should be determined based upon the depth of contaminated soil associated with the dose model used to generate surface soil DCGLs. If subsurface residual radioactivity is present, dose modeling may be conducted for both surface and subsurface soils and DCGLs developed for each. In these cases, the MARSSIM methodology will need to be supplemented or an alternative methodology will need to be developed to demonstrate compliance with radiological criteria for license termination.

Through the FSS, the licensee would have to demonstrate that the $DCGL_W$ is satisfied within the specified exposure area for each survey unit. The licensee should consider the area assumed in the dose modeling for consistency against the area of the survey unit and the area over which concentrations are averaged for comparison against DCGLs. The licensee should also develop $DCGL_{EMC}$ values for smaller areas within the larger area of residual radioactivity as part of the FSS design (e.g., develop $DCGL_{EMC}$ values for smaller areas in the survey unit between soil sampling locations that may be missed by sampling). If it is certain that the residual radionuclide concentration is limited to a specific lateral extent, the licensee may also incorporate the "area of residual radioactivity" into the identification of DCGLs. Computer modeling codes, such as RESRAD-ONSITE, allow the user to directly specify the area of residual radioactivity and adjust the dose as a function of the specified area in a unique manner for each exposure pathway.

The NRC recommends that the licensee use dose modeling to calculate $DCGL_{EMC}$s. For example, it can use RESRAD-ONSITE and RESRAD-BUILD to calculate dose for the larger area of concern (i.e., the entire contaminated area) and the dose from the smaller areas of potentially elevated concentration, separately. It can then use dose modeling results to calculate the cleanup levels for the larger area of concern and for the smaller elevated areas ($DCGL_W$ and $DCGL_{EMC}$s, respectively). If only a single elevated area is present, the EMC is acceptable if Equation 1 is met (modified from MARSSIM, Equation 8.2)

$$\frac{\delta}{DCGL_W} + \frac{average\ concentration\ in\ the\ elevated\ area - \delta}{DCGL_{EMC}} < 1 \qquad \text{Equation 1}$$

Where $\delta$ = the average residual radioactivity concentration for all sample points in the survey unit.

In cases where there is more than one elevated area, a separate term should be included in the calculation for each area of elevated activity. The unity rule is satisfied when radionuclide mixtures yield a combined fractional concentration limit that is less than or equal to one. In situations where there is more than one radionuclide at a single source (elevated area or "hot spot"), the sum of the individual ratios also cannot be greater than or equal to 1. The sum of fractions rule applies in situations where there are multiple radionuclides and sources. For example, consider a site where residual radioactivity is present in two elevated areas or hot spots, areas A and B, as well as in a larger area of concern. The concentrations of radionuclides 1, 2, and 3, in elevated or hot spot areas A and B, and the larger area, W, must all be considered and the sum of fractions must be less than or equal to 1 (see Equation 2). Note that the general area concentrations $C_{xW}$ for each radionuclide, x=1, 2, or 3, can be subtracted

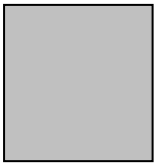
from the elevated area concentrations, $C_{XA}$ and $C_{XB}$ for each radionuclide, in Equation 2 consistent with Equation 1.

$$\frac{C_{1A}}{DCGL_{1A}} + \frac{C_{2A}}{DCGL_{2A}} + \frac{C_{3A}}{DCGL_{3A}} + \frac{C_{1B}}{DCGL_{1B}} + \frac{C_{2B}}{DCGL_{2B}} + \frac{C_{3B}}{DCGL_{3B}} + \frac{C_{1W}}{DCGL_{1W}} + \frac{C_{2W}}{DCGL_{2W}} + \frac{C_{3W}}{DCGL_{3W}} \leq 1 \qquad \text{Equation 2}$$

The following examples present different options for considering the contributions of multiple elevated areas within a single larger area. In general, two or more smaller areas modeled independently and combined will result in a higher dose then if contaminant concentrations are averaged across a single area. Although the higher dose may be considered conservative, it is often unrealistic, because it may assume, for example, that an individual spends all of his or her time in multiple locations simultaneously. The examples below demonstrate how the assumptions made about the contamination on a site can affect whether or not the site passes the cleanup criteria.

Example 1—Base Case

For this example, a survey unit of 10,000 m$^2$ is uniformly contaminated with americium (Am)-241 at a concentration of 9 pCi/g. The DCGL is 10 pCi/g. Because the concentration (9 pCi/g) is below the DCGL, the survey unit will pass.

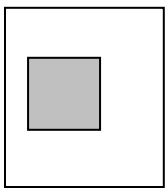


- A 10,000 m$^2$ survey unit with uniform Am-241 contamination (no elevated areas)
- Am-241 concentration = 9 pCi/g
- $DCGL_W$ = 10 pCi/g

$$\frac{9\frac{pCi}{g}}{10\frac{pCi}{g}} = 0.9 \qquad 0.9 < 1.0 \rightarrow \text{Survey unit passes cleanup criteria}$$

Example 2—Single Elevated Area in the Survey Unit

Example 2 considers a 10,000 m$^2$ survey unit with a single 300 m$^2$ elevated area contaminated with Am-241 at a concentration of 9 pCi/g. The $DCGL_W$ for the site is 10 pCi/g and the $DCGL_{EMC}$ for the elevated area (300 m$^2$) is 12. As expected, the smaller contaminated area within the larger survey unit results in a smaller sum of fraction (0.75), which is less than 1. Therefore, the survey unit with these characteristics also passes the cleanup criteria.

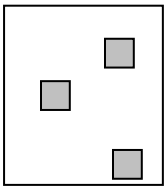


- A 10,000 m$^2$ survey unit with no contamination except for a 300 m$^2$ elevated area
- Am-241 concentration = 9 pCi/g
- $DCGL_W$ = 10 pCi/g
- $DCGL_{EMC}$ = 12

$$\frac{9\frac{pCi}{g}}{12\frac{pCi}{g}} = 0.75$$ 0.75 < 1.0 → Survey unit passes cleanup criteria

Example 3—Multiple Elevated Areas or Hot Spots in a Single Survey Unit

As shown in Example 3, evaluating multiple smaller elevated areas often results in a higher dose than if the areas were combined and modeled as a single area. Instead of the single 300 $m^2$ elevated area used in Example 2, Example 3 evaluates three 100 $m^2$ elevated areas within the same 10,000 $m^2$ survey unit. The $DCGL_W$ for the site is still 10 pCi/g, and the $DCGL_{EMC}$ (100 $m^2$) is 18.6 pCi/g. While if a single 100 $m^2$ elevated area was present, the dose criterion could be easily met, summing the dose contributions of the three individual elevated areas together results in a value greater than 1 and therefore, exceeds the cleanup criteria.



- A 10,000 $m^2$ survey unit with no contamination except for 3 X 100 $m^2$ elevated areas
- Am-241 concentration = 9 pCi/g
- $DCGL_W$ = 10 pCi/g
- $DCGL_{EMC}$ = 18.6

A 100 $m^2$ elevated area:

$$\frac{9\frac{pCi}{g}}{18.6\frac{pCi}{g}} = 0.48$$

Combining the results of each of the three 100 $m^2$ elevated areas:

$$\frac{9\frac{pCi}{g}}{18.6\frac{pCi}{g}} + \frac{9\frac{pCi}{g}}{18.6\frac{pCi}{g}} + \frac{9\frac{pCi}{g}}{18.6\frac{pCi}{g}} = 1.45$$

In this case the higher dose is unrealistic, because it assumes that a receptor spends all of his or her time on each contaminated area simultaneously. The licensee can determine a more reasonable estimate of potential dose by combining the individual elevated areas into a single larger elevated area within the survey unit, as in Example 2.

In addition to specifying a limited area of residual radioactivity in developing the site-specific DCGLs for soil, the licensee should appropriately represent the vertical extent of residual radioactivity within the area. The screening DCGLs and the DandD code assume that residual radioactivity is contained within the uppermost 15 cm of soil. If the licensee intends to leave residual radioactivity at depths below approximately 15 cm, the calculation of the $DCGL_W$ should

reflect a greater thickness of residual radioactivity. Otherwise, leaving residual radioactivity below 15 cm may not be acceptable.

For subsurface residual radioactivity (i.e., residual radioactivity at depths greater than approximately 15 cm), the NRC staff should evaluate whether the licensee has reviewed existing historical site data (including previous processes or practices) and site characterization data to establish an adequate conceptual model of the subsurface source specifically about the horizontal and vertical extent of residual radioactivity. The licensee should evaluate lateral and vertical trends of variation in concentration for each specific radionuclide. Because certain radionuclides have higher mobility than others, radionuclide ratios may not be maintained as constant across subsurface soil. In other words, radionuclide concentrations within the unsaturated zone may vary, depending on the original source location and the time since the source existed. The NRC staff should evaluate whether the licensee has evaluated the physical and chemical properties of the source and the attenuating properties of the subsurface materials to assess the potential for radionuclide leaching and transport. In this context, the reviewer should evaluate the selected physical parameters and conceptual model of the site against the actual subsurface hydrostratigraphy to evaluate the acceptability of the parameter selections if found to be important to dose. The reviewer should also consider (1) heterogeneity in subsurface soils, and (2) the depth to the water table if found to be important to dose.

If the thickness of residual radioactivity that will remain at a site is generally uniform across the site, the licensee may choose to use an upper bounding value for modeling the thickness. The NRC reviewer should evaluate the representative thickness value proposed by the licensee to ensure that the value selected does not lead to a significant underestimate of dose, particularly for sites where the vadose thickness is quite variable. For example, given the variable depth to the water table, a licensee may propose to use an area-weighted approach to assign the vadose zone thickness in dose modeling. However, if the timing of peak dose from groundwater dependent pathways is important to the compliance demonstration, then representation of the area with a thinner vadose zone may be important, for example.

If appropriate, the licensee should provide maps and cross sections detailing the proposed lateral and vertical extent of residual radioactivity left on the site.

#### *I.2.3.2 Dose Modeling Approach Two: Assess Dose Based on Actual Concentrations*

An alternative objective that a licensee may have for performing and submitting dose modeling may be to assess doses attributable to specific quantities of radioactive material. Although the development of DCGLs focuses on the determination of radionuclide concentrations corresponding to a specified dose, the dose assessment objective focuses on the determination of doses corresponding to specified radionuclide concentrations.

In this situation, the licensee should give much more attention to the source abstraction and address all elements of the source term abstraction:

- identify the radionuclides of concern
- delineate the spatial extent of residual radioactivity
- represent the spatial variability of residual radioactivity
- incorporate the spatial variability of physical and chemical characteristics of the contaminated media

The licensee should focus on the distribution of radioactive material expected to be present at the time of FSS and subsequent site release. The licensee may assess doses attributable to existing radiological conditions at the site, if it can demonstrate that the existing radiological conditions reasonably bound conditions expected at the FSS, from a dose perspective.

The first two elements of source abstraction—radionuclides of concern and spatial extent—were considered in the discussion of source abstraction for the development of DCGLs. Spatial variability was not considered, since it is statistically evaluated as part of the FSS. If the dose modeling approach using actual radionuclide concentrations is used, however, spatial variability should be factored into the source abstraction before modeling.

Assuming that the licensee has identified the radionuclides of concern and delineated the spatial extent of residual radioactivity, it should project the residual radionuclide concentration distribution and total residual radionuclide inventory across the site. The licensee should tie this projection directly to the characterization of existing radiological conditions at the site. The site may then be divided into relatively large areas that are radiologically distinct, based on radionuclide concentration or depth of residual radioactivity. The licensee should statistically demonstrate that the radionuclide concentrations or depth within an area may be relatively uniform, taking into account the spatial distribution of the data. Similarly, within the larger areas, the licensee should statistically delineate relatively small areas of projected elevated radionuclide concentrations or increased depth. (The licensee should discuss the reason for leaving the elevated concentrations in place as residual radioactivity.)

When complete, the licensee's source term abstraction should define a site divided into relatively large areas of statistically uniform radionuclide concentrations and residual radioactivity depth. Within these areas may be relatively small areas of elevated concentration or increased depth. Assuming that the physical and chemical conditions across the site are relatively uniform, the licensee may use this source abstraction for modeling and proceed with the dose assessment.

The following is a suggested approach:

- Consider each relatively large area independently and initially ignore the relatively small elevated areas within each large area.

- Assess dose based on the properties of a large area, taking the areal extent into account.

- Repeat the dose assessment but assume an essentially infinite areal extent. The specific approach will depend on the computer modeling code used. This should quantify the impact of dividing the site into artificial modeling areas.

- Assess dose attributable to each limited area of elevated concentration, assuming no residual radioactivity exists outside the limited area. This may then be combined with the dose attributable to the surrounding larger area, to assess the impact of leaving the elevated concentrations.

In some cases, it may not be practical to separate a site into areas with relatively uniform radionuclide concentrations; sometimes areas to be evaluated will have nonuniform distributions of concentrations. In such cases, for performing the second step above, there may be a question about what statistical value best represents the radionuclide concentration for the large

area. Log-normal distributions occur frequently in nature and are not unexpected when surveying contaminated sites. For log-normal distributions, the geometric mean is often used as a descriptor of the distribution. However, the geometric mean concentration should not be used as the average value for the source concentration for dose calculations, as use of the geometric mean could lead to a significant underestimate of the dose compared to use of the arithmetic mean. Because (1) the dose rate is proportional to radionuclide concentration, (2) it is reasonable to assume that the receptor spends an equal amount of time in each area of the site, and (3) each characterization data point represents an equal area, use of the arithmetic mean (and not the geometric mean) is more technically defensible for calculating average concentrations for use in dose modeling (second step above). If samples are not taken randomly or systematically (and thus data points represent unequal areas), weighted means may be appropriate, with application of weighting factors consistent with the assumptions of receptor exposures.

The above discussion does not specifically address the determination of relatively significant large or small areas. This designation will depend on the areal assumptions underlying the computer modeling code used. For example, the DandD code considers the area of cultivation to be uniformly contaminated and irrigated. The area of cultivation depends on the cultivation requirements defined by the specific exposure scenario. While DandD allows the user to specify either (1) an "unlimited" area of contamination consistent with the underlying conceptual model, or (2) smaller areas of contamination, the relationship between area and dose is defined in a simplistic manner. Furthermore, the area of contamination is not treated as an uncertain parameter and, therefore, an automated sensitivity analysis on the area of contamination is not available. With respect to the RESRAD family of codes, ingestion pathway doses are largely dependent on the area of the source; however, the impact of area on dose varies depending on the specific pathway being considered. For example, soil and plant ingestion doses scale directly with areas up to 1,000 $m^2$,[5] while animal pathway doses scale directly with areas up to 20,000 $m^2$. The licensee should discuss and justify the designation of relatively large and relatively small areas, based on the computer code used. The licensee can provide additional information (e.g., results of alternative scenarios evaluating the sensitivity of the dose modeling results to area) to lend more support for the compliance demonstration.

As discussed above, the licensee may also have to consider the impact of multiple areas of elevated concentration within a single larger area. In general, modeling two small areas independently and combining the results of the two dose assessments should result in a higher dose than if the two areas were combined and modeled as a single area (see Examples 1–3 in Section I.2.3.1). The higher dose may be unrealistic if it assumes that the receptor location relative to each contaminated area is such that the dose is maximized from each contaminated area independently. For a more reasonable estimate of potential dose, these smaller areas may be combined into a single larger area, if the concentrations within the smaller areas are comparable (e.g., see Example 3 in Section I.2.3.1). If this is not the case, then the licensee may model each smaller area individually and modify the exposure scenario and critical group assumptions for each area (e.g., time spent on each area) and combine the results.

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5 When a value of "-1" is input into the field for the contaminated fraction for plant food, and the size of the contaminated zone is equal to or greater than 1,000 $m^2$, RESRAD-ONSITE and RESRAD-OFFSITE assume that 50 percent of the crops consumed by the receptor come from a garden grown in contaminated soil (i.e., no more than 50 percent of the produce comes from the contaminated garden and 2,000 $m^2$ is needed to support 100-percent home grown produce ingestion rates). For areas less than 1,000 $m^2$, RESRAD-ONSITE and RESRAD-OFFSITE linearly scale the consumption rates of contaminated produce down from 50 percent for 1,000 $m^2$ areas to 0 percent for 0 $m^2$ areas.

The example illustrated in Figure I.1 presents an acceptable method for considering the contributions of multiple elevated areas or "hot spots" within a larger contaminated area, when the concentrations of the elevated areas are variable with each other, as well as with the larger area of residual radioactivity. Consider a site with two relatively small areas of elevated radioactivity in comparison to levels of radioactivity over a much larger area. Because most dose modeling codes, including DandD and RESRAD-ONSITE, assume that the receptor is located in the center of the contaminated area, the "conceptual model" in Figure I.1 depicts overlapping contaminated areas. However, to avoid overestimating the dose when the contaminated areas are overlaid and summed, the larger contaminated area is run first with an average concentration of 3 pCi/g. Next, the second most elevated area or "hot spot," which has an average concentration of 10 pCi/g, is simulated with a concentration of 3 pCi/g less than the average concentration of 10 pCi/g (or with a value of 7 pCi/g), so as not to double-count the activity in the larger area to which it is summed. The second most elevated area or hot spot is assumed to be 200 $m^2$ instead of 100 $m^2$, to account for the fact that 100 $m^2$ of the simulated contaminated area overlaps the most elevated 100 $m^2$ area that is considered separately. Using this approach, the second most elevated area or hot spot with an average concentration of 10 pCi/g is conceptually assumed to surround the most elevated 100 $m^2$ area, (see yellow doughnut surrounding the orange, most elevated, hot spot in Figure I.1). This approach is acceptable, as it is impossible for the receptor to occupy two different 100 $m^2$ areas at the same time, and assuming that the receptor is located in the center of the hottest area on site for the entire exposure period is conservative. Finally, the most elevated area or hot spot, which has an average concentration of 16 pCi/g, is simulated with a concentration 10 pCi/g less than the average concentration of 16 pCi/g (or with a value of 6 pCi/g), so as not to double-count the activity assigned in the larger area simulation of 3 pCi/g, and not double-count the activity assigned in the second most elevated area simulation of 7 pCi/g. Although the geometry and locations of the elevated areas or "hot spots" differ in the "conceptual model" versus the "actual" configuration depicted in Figure I.1, the assumed geometry and elevated area location tends to overestimate the dose with the receptor standing directly on top of the hottest contaminated area on the site and in relatively close proximity to the second most contaminated area on site. Depending on the actual size and geometry of the elevated areas being simulated, this method may produce overly conservative results. If less conservative methods are needed to demonstrate compliance, the licensee may propose alternative methods that will require approval by NRC reviewers on a case-by-case basis.

Figure I.2 illustrates how vertical heterogeneity can be considered in dose modeling. In the example, variability in concentration for the first 5 cm of soil and the next 10 cm of soil is considered in the dose modeling. In the example, two simulations are run and the doses from the simulations added together. In the first simulation, the entire 15 cm of soil is considered but using the lower concentration of the deeper 5 to 15 cm soil interval. In the second simulation, the top 5 cm of soil is considered using the difference in concentration between the shallower 0 to 5 cm and the deeper 5 to 15 cm soil intervals. The doses from the two simulations are summed to provide a dose estimate for the two soil intervals represented in the model.

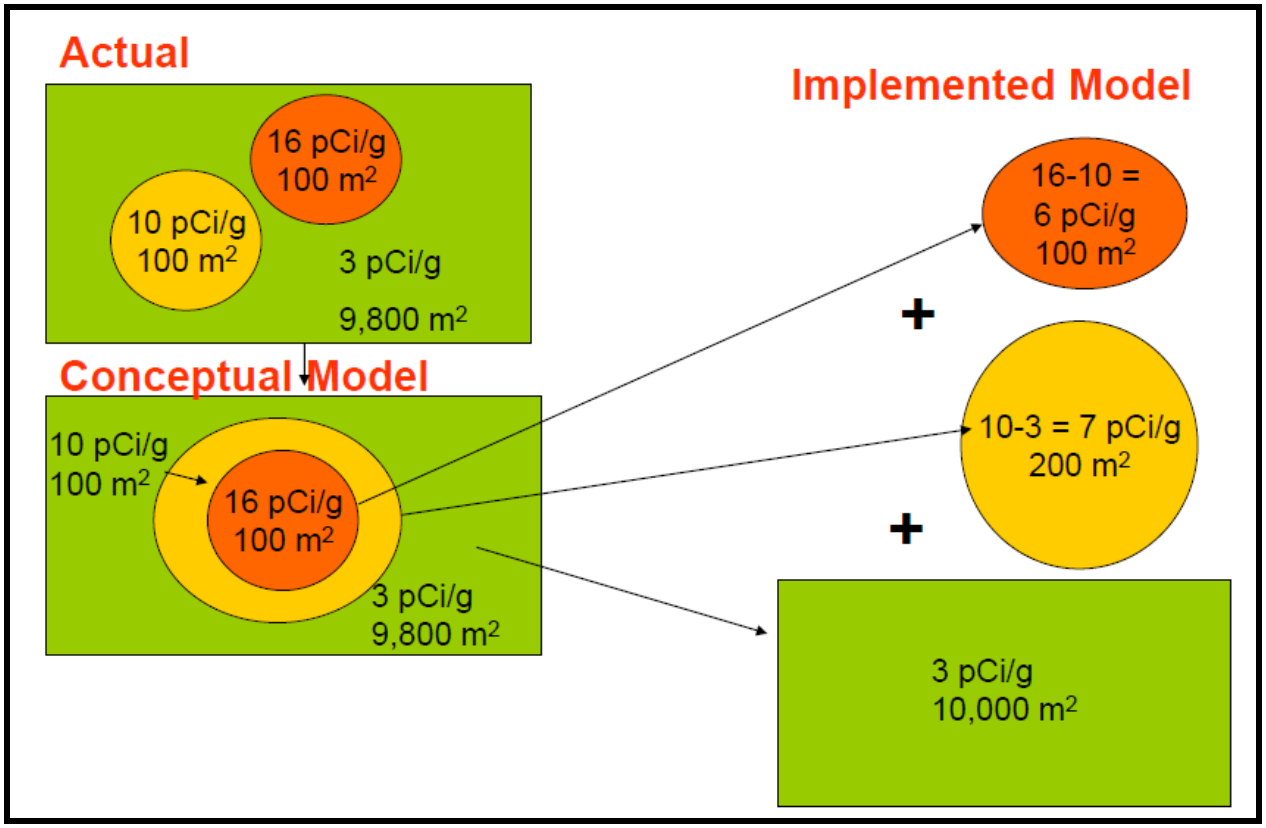


**Figure I.1 Approach to Considering Multiple Elevated Areas or Hot Spots**

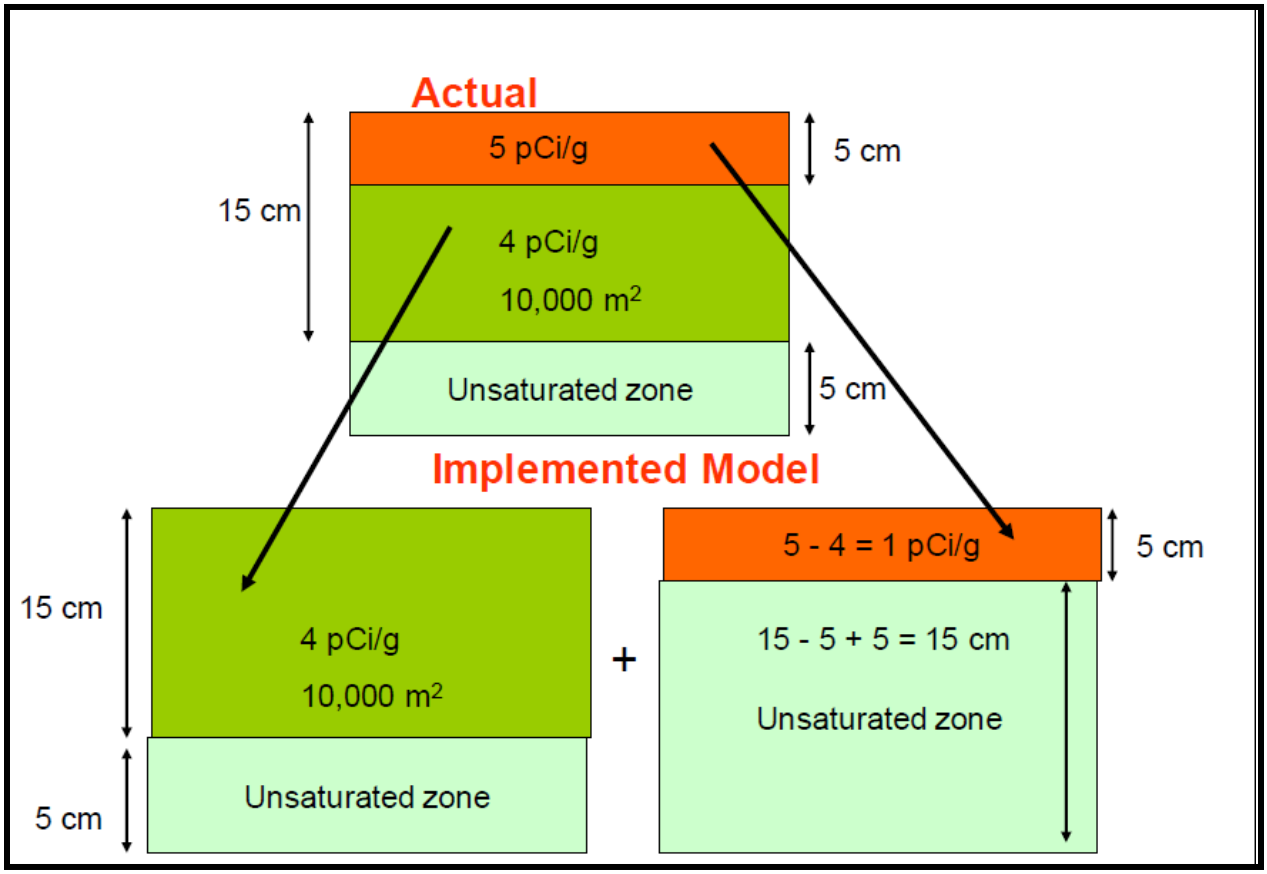


**Figure I.2 Approach to Considering Vertical Heterogeneity**

**Case Study 1: Evaluation of the Impact of Variability of Exposure Area Concentrations**

Using the dose modeling approach, analysts can evaluate the uncertainty in dose modeling predictions based on spatial variability in exposure area concentrations using a code such as SADA. A case study by the NRC staff used soil sampling data to calculate average concentrations for various exposure areas at a thorium-contaminated site. The default exposure area first considered was a 2,000 $m^2$ area, which is of sufficient size to support the assumed external exposure pathway occupancy factors for a resident gardener, as well as the assumed plant ingestion rates derived from regional data. The NRC used SADA to calculate a spatially moving average concentration or the possible set of concentrations for 2,000 $m^2$ exposure areas. The NRC then used the distribution of 2,000 $m^2$ exposure area concentrations for comparison against an area-wide DCGL derived from dose modeling. The staff also performed sensitivity analyses that considered a range of exposure area sizes to provide additional information with respect to other types of uncertainty, such as institutional, land use, or behavioral uncertainties (e.g., uncertainty in future land use, including the size and location of parcels of land a resident receptor may reside on following license termination and redevelopment of the released site). It is important to note that, as the size of the exposure area changes, so may the exposure pathways, occupancy factors, and other exposure scenario-related parameters. For example, if one were to consider relatively small exposure areas appropriate for an apartment or condominium resident, which will result in higher maximum exposure area concentrations, then it may be (1) unreasonable to assume certain pathways are viable, such as homegrown produce ingestion, and (2) shielding factors for the apartment or condominium residence may need to be adjusted to more accurately calculate external pathway dose. In these cases, dose modeling can estimate dose for the smaller exposure areas using more representative parameters and pathways appropriate for the exposure scenario being considered, with appropriate justification. Care should also be taken to ensure that the statistics are not skewed due to the presence of large areas of unimpacted soils (e.g., only Class 1–3 areas should be included in the analysis for particularly large sites with significant portions of the site that are not radiologically affected by facility operations).

Evaluation of uncertainty or variability in exposure area concentrations will lead to a more comprehensive evaluation of risk to the average member of the critical group. Analysis of a range of exposure areas, including smaller exposure areas that may be appropriate for alternative exposure scenarios, will better inform the decisionmaker with respect to less likely but plausible exposure scenarios that may be more limiting. As the size of the exposure areas decreases and approaches the size of true elevated areas or hot spots, SADA can also provide valuable information on the distribution of elevated area concentrations at a site. Consideration could then be given to the number and location of elevated areas on the site to ensure that potential risk is well understood. In some cases, it may be reasonable to assume the receptor spends his time at only one elevated area or to consider some form of time or spatial weighting if the elevated areas are located a considerable distance apart and the exposure area is rather large in relation to the elevated areas.

## I.3 Criteria for Selecting and Modifying Exposure Scenarios, Pathways, and Critical Groups

### I.3.1 Introduction

After the source term has been evaluated, the question becomes, "How could humans be exposed either directly or indirectly to residual radioactivity?" or "What is the appropriate exposure scenario?" Each exposure scenario should address the following questions:

(1) How does the residual radioactivity move through the environment?

(2) Where can humans be exposed to the environmental concentrations?

(3) What is the likely land use(s) in the future for these areas?

(4) What are the exposure group's habits that will determine exposure? (For example, what do they eat and where does it come from? How much? Where do they get water and how much? How much time do they spend on various activities?)

The ultimate goal of dose modeling is to estimate the dose to a specific receptor. Broad generalizations of the direct or indirect interaction of the affected receptors with the residual radioactivity can be identified for ease of discussion among the licensee, regulator, public, and other interested parties. Exposure scenarios are defined as reasonable sets of activities related to the future use of the site. Therefore, exposure scenarios describe future land uses, human activities, and the behavior of the natural system.

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

By combining knowledge about the answers to Questions 1 and 2, the licensee can develop exposure pathways. These are the routes that residual radioactivity travels through the environment, from its source, until it interacts with a human. They can be fairly simple (e.g., surface-soil residual radioactivity emits gamma radiation, which results in direct exposure to the individual standing on the soil) or they can be fairly involved (e.g., the residual radioactivity in the surface soil leaches through the unsaturated soil layers into the underlying aquifer, and the water from the aquifer is pumped out by the exposed individual for use as drinking water, which results in the exposed individual ingesting the environmental concentrations). Exposure pathways typically fall into three principal categories, identified by the manner in which the exposed individual interacts with the environmental concentrations resulting from the residual radioactivity: ingestion, inhalation, or external (i.e., direct) exposure pathways.

As required under Subpart E, the licensee evaluates the dose from residual radioactivity for the average member of the critical group, which is not necessarily the same as the maximally exposed individual. This is not a reduction in the level of protection provided to the public but an attempt to emphasize the uncertainty and assumptions needed in calculating potential future doses, while limiting boundless speculation on possible future exposure scenarios. Although it

is possible to actually identify with confidence the most exposed member of the public in some operational situations (e.g., through monitoring, time studies, distance from the facility), identification of the specific individual who may receive the highest dose some time (up to 1,000 years) in the future 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 the statistical results from data sets (e.g., the breathing rate is based on the range of possible breathing rates) gathered from groups of individuals. Although bounding assumptions could be used to select values for each of the parameters (e.g., the maximum amount of meat, milk, vegetables, possible exposure time), the result could be an extremely conservative calculation of an unrealistic exposure scenario and may lead to excessively low allowable residual radioactivity levels, compared to the actual risk.

Calculating the dose to the critical group is intended to bound the individual dose to other possible exposure groups, because the critical group is a relatively small group of individuals, who, because of their habits, actions, and characteristics, could receive among the highest potential doses at some time in the future. By using the hypothetical critical group as the dose receptor, coupled with prudently conservative models, it is highly unlikely that any individual would actually receive doses in excess of that calculated for the average member of the critical group. The licensee should base the description of a critical group's habits, actions, and characteristics on credible assumptions, and the information or data ranges used to support the assumptions should be limited in scope to reduce the possibility of adding members of less exposed groups to the critical group.

As low as is reasonably achievable (ALARA) analyses should use the dose based on the reasonably foreseeable land use for any cost-benefit calculations performed.

### I.3.2 Issues in Selecting and Modifying Exposure Scenarios, Pathways, and Critical Groups

The definition of exposure scenarios, identification of a critical group with its associated exposure pathways, and the dose assessment based on that definition can be generic or site-specific. Licensees might do the following:

- Use screening exposure scenarios, screening groups, and pathway parameters as described in NUREG-1549 (NRC, 1998d) and the NUREG/CR-5512 series, "Residual Radioactive Contamination from Decommissioning," Volumes 3 and 4, issued October 1999 (e.g., NRC, 1999c, 1999d). This can be used for either screening or site-specific analyses.

- Use the default screening exposure scenarios as a starting point to develop more site-specific pathway analyses or critical group habits.

- Develop site-specific exposure scenarios and critical groups and identify associated exposure pathways from scratch.

To establish site-specific exposure scenarios, critical groups, or sets of exposure pathways, the licensee may need to justify its selections. For some licensees, this may require minimum amounts of site-specific data to support the assumptions inherent in the existing default screening exposure scenarios or for removing specific exposure pathways. For others, the licensee may need to thoroughly investigate and justify the appropriateness of the selected

exposure scenarios or critical groups, which may include evaluation of alternative exposure scenarios or critical groups. If a licensee creates the exposure scenario and associated critical group based on site-specific conditions (e.g., at a site that is grossly different than the assumptions inherent in the default exposure scenarios), it should include documentation that provides a transparent and traceable audit trail for each of the assumptions used in developing the exposure scenario and critical group (e.g., justify the inclusion (or exclusion) of a particular exposure pathway).

### I.3.3 Recommended Approaches

#### *I.3.3.1 Screening Analyses*

In the case of screening, the decisions involved in identifying the appropriate exposure scenario and critical group, with their corresponding exposure pathways, have already been made. Exposure scenario descriptions acceptable to the NRC staff for use in generic screening are developed and contained in NUREG/CR–5512, Volume 1. NUREG/CR–5512, Volume 3, and NUREG-1549 provide the rationale for applicability of the generic exposure scenarios, critical groups, and pathways at a site; the rationale and assumptions for exposure scenarios and pathways included (and excluded); and the associated parameter values or ranges (NUREG/CR-5512, Volume 3, contains detailed information on data to support development of parameter distributions). A description of the screening exposure scenarios and associated pathways is provided in Table I.2.

#### *I.3.3.2 Site-Specific Analyses*

Site-specific analyses give licensees greater flexibility in developing the compliance exposure scenario. The licensee should justify its selection of this scenario based on reasonably foreseeable land use at the site. This scenario should result in an exposure to the public, such that no other exposure scenario, using reasonably foreseeable land use assumptions, will result in higher doses to its exposure group(s). The level of justification and analysis provided by the licensee will depend on how much credit is taken by adding “realism” to the exposure scenario based on site-specific information (i.e., how much credit is being taken for elimination of pathways or reduction in pathway contributions relative to more conservative assumptions made in screening exposure scenarios). As the analysis becomes more realistic, greater degrees of justification and, potentially ancillary analyses, will be required. For example, a site is currently zoned as industrial, and the local area is a mix of suburban, commercial, and industrial uses. Rural uses of the property are less likely but plausible for the foreseeable future. If it chose to use the generic screening exposure scenario, the licensee would need to provide limited justification. If the licensee proposed to use a maintenance worker exposure scenario assuming industrial land use as the compliance exposure scenario, it would need to provide quantitative analyses of, or a qualitative argument discounting the need to analyze, other competing exposure scenarios (based on industrial land use and on suburban or commercial land use) to justify the selection of the compliance exposure scenario. In addition, the licensee would need to provide analyses of the rural use of the land to show what impacts would occur from the less likely but plausible exposure scenario.

Site-specific analyses can use the generic screening exposure scenario(s) with little justification. The licensee may need to justify that the site contains neither physical features nor locations of residual radioactivity (other than those assumed in the screening analyses), which would invalidate the assumptions made in developing the exposure scenarios. If site or source features are found to be incompatible with exposure scenario assumptions, the licensee should

justify why the generic exposure scenario are nonetheless appropriate for use in the dose modeling. A site can fail to meet the requirements of the conceptual model (see Section I.4 of this appendix) without invalidating the generic exposure scenario, and situations can arise where the default exposure scenario is no longer the limiting case. For example, the site may have preexisting groundwater contamination, which is counter to the assumptions in the conceptual model inherent in the screening models. However, this may not require any change in the exposure scenario, because the residential farmer scenario may still be an appropriate scenario, as it contains all of the appropriate exposure pathways, including groundwater use for drinking, irrigation, and for animals. Alternately, if the residual radioactivity were a volumetric source in the walls of a building, rather than on the building surfaces, the generic exposure scenario of an office worker may not be the exposure scenario leading to the critical group. For certain sets of radionuclides, a building renovation exposure scenario may be more limiting because of the exposure to the airborne concentration of material as the walls are disturbed.

**Table I.2 Pathways for Generic Exposure Scenarios**

| |
|---|
| **Building Occupancy Exposure Scenario**<br>This exposure scenario accounts for exposure to fixed and removable residual radioactivity on the walls, floor, and ceiling of a decommissioned facility. It assumes that the building may be used for commercial or light industrial activities (e.g., an office building or warehouse).<br><br>Pathways include the following:<br><br>• external exposure from building surfaces<br>• inhalation of (re)suspended removable residual radioactivity<br>• inadvertent ingestion of removable residual radioactivity |
| **Resident Farmer Exposure Scenario**<br>This exposure scenario accounts for exposure involving residual radioactivity that is initially in the surficial soil. A farmer moves onto the site and grows some of his or her diet and uses water tapped from the aquifer under the site.<br><br>Pathways include the following:<br><br>• external exposure from soil<br>• inhalation to (re)suspended soil<br>• ingestion of soil<br>• ingestion of drinking water from aquifer<br>• ingestion of plant products grown in contaminated soil and using aquifer to supply irrigation needs<br>• ingestion of animal products grown on site (using feed and water derived from potentially contaminated sources)<br>• ingestion of fish from a pond filled with water from the aquifer |

The licensee can develop site-specific exposure scenarios, critical groups, and pathways for any situation. Some cases where changes to the default exposure scenarios or modification to exposure pathways are or are likely to be appropriate include the following:

(1) Major pathways (e.g., the groundwater pathway or agricultural pathways) associated with the default screening scenarios could be eliminated, for either physical or site-use reasons.

(2) The location of the residual radioactivity and the physical features of the site are outside the major assumptions used in defining the default critical group and exposure scenarios.

(3) Restricted use is proposed for a site.

The second case listed above can be ambiguous, as a number of assumptions key to the development of the DandD screening tool do not affect the exposure scenario description, and the NRC reviewer may need to evaluate whether the initial generic exposure scenario would still be appropriate for the site.

Modifying exposure scenarios or developing a site-specific critical group requires information on plausible uses of the site and demographics. Such information might include considerations of the prevailing (and future) uses of the land and physical characteristics of the site that may constrain site use. The licensee should categorize potential land uses as reasonably foreseeable, less likely but plausible, or implausible. Any land uses that similar property in the region currently has, or may have in the near future (e.g., approximately 100 years), should be characterized as reasonably foreseeable. The licensee should consider trends and area land use plans in determining the likelihood of potential land use. Land uses that are plausible, generally because similar land was historically used for similar purposes, could be characterized as less likely but plausible if found to be counter to the current trends or regional experience, (e.g., rural use of property currently in an urban setting). Implausible land uses are those that, because of physical limitations, could not occur (e.g., residential land use for an underwater plot of land). It may be necessary to evaluate several potential critical groups, based on different combinations of site-specific exposure scenarios developed from expected land use, pathways, and demographics, to determine the group receiving the highest exposure.

Depending on the resulting exposure scenarios, considerations of offsite exposure by either transport (e.g., through groundwater) or material transfer may be necessary to identify the critical group. Thus, the licensee should consider if offsite uses are reasonably foreseeable. If offsite uses are found to be reasonably foreseeable, such offsite uses should be analyzed to determine if the offsite user receives a higher dose compared to an onsite user and if offsite users should be identified as the critical group.

Similar considerations apply for restricted release. Thus, when analyzing the dose under restricted conditions, the nature of the critical group is likely to change because of these restrictions and controls. Site restrictions and institutional controls can restrict certain kinds of activities and land or water uses associated with the physical features of the site. The detailed definition of the exposure scenarios considered for restricted release need to include the impact of the control provisions on the location and behavior of the average member of the appropriate critical group.

For restricted use, licensees must also evaluate doses assuming institutional controls are no longer in effect. This evaluation should address (1) the associated degradation of engineered barriers assuming there is no active maintenance, and (2) exposure scenarios assuming a loss of institutional controls immediately following license termination (i.e., time=0 years). Section 3.5 of this volume contains additional information on modeling the performance and degradation of engineered barriers in dose modeling analyses.

The NRC license reviewer should evaluate the licensee's justifications for its exposure scenarios using the following appropriate guidance. The guidance is characterized by the general approach used in developing the exposure scenarios, either (1) modifying existing generic exposure scenarios or (2) developing site-specific scenarios from "scratch."

**MODIFICATION OF GENERIC EXPOSURE SCENARIOS**

First, the NRC license reviewer should evaluate whether the generic exposure scenario was applicable to the site before the licensee started modifying the exposure scenario, based on physical features or restrictions, and should identify the modifications and evaluate the licensee's justification for those changes. Table I.3 lists some common exposure scenarios but is by no means comprehensive. The Sandia Letter Report, "Process for Developing Alternate Scenarios at NRC Sites Involved in D&D and License Termination," issued January 2000 (Thomas, et al., 2000)), which is included, in part, as Appendix M in this volume, provides information to assist a licensee or reviewer with respect to the modification of default exposure scenarios using site-specific information. Specific guidance on acceptable justifications for modifying the default exposure scenarios is provided below, based on different types of site-specific information. Additionally, if the licensee's intent is restricted release, the NRC should review the final exposure scenario for the case where restrictions are in place. Based on either site restrictions or site-specific data, the licensee's justifications should support the elimination from the analysis of exposure scenarios and pathways. The NRC should focus the review on the most risk-significant pathways and model components.

**Table I.3 Potential Exposure Scenarios for Use in Dose Assessments**

- building occupancy (generic screening—based on NUREG/CR-5512)
- residential farmer (generic screening—based on NUREG/CR-5512)
- urban construction (contaminated soil, no suburban or agricultural uses), meant for small urban or industrial sites cleared of all original buildings, with only contaminated land and/or buried waste remaining
- residential (residential farmer exposure scenario with eliminated exposure pathways appropriate for those urban or suburban sites where farming is not a realistic projected future use of the land)
- recreational user (where the site is preserved for recreational uses only)
- maintenance worker (tied to the recreational user exposure scenario but involves the grounds keepers maintaining or building on the site)
- hybrid industrial building occupancy (adds contaminated soil, while building may or may not be contaminated)
- offsite drinking water (e.g., no onsite use of groundwater; offsite impacts from the contaminated plume)

The licensee may need to evaluate whether the final modified exposure scenario is still the limiting reasonable representation of the critical group at the site. This may involve investigating exposure pathways not covered in the default exposure scenarios.

## DEVELOPMENT OF ALTERNATIVE EXPOSURE SCENARIOS

In some decommissioning cases, either the location of the residual radioactivity, the physical characteristics of the site, or planned institutional restrictions may make the default exposure scenarios inappropriate. In other cases, the licensee may wish to provide a transparent and traceable development of the compliance and other exposure scenarios, starting with the potential land use and the site conditions. Development (and review) of alternative exposure scenarios may involve iterative steps to create the conceptual model of the site. For example, the licensee may (1) develop a generic list of exposure pathways, (2) develop the site conceptual model to screen the generic list, (3) aggregate or reduce the remaining exposure pathways to the major exposure pathways, and (4) reevaluate the conceptual model to verify that all the necessary processes are included.

A brief summary of the NRC–recommended pathway analysis process follows. Appendix K contains an example of exposure scenarios developed for PSR.

- Compile a list of exposure pathways applicable to any contaminated site. A number of existing sources of information can be used, for example, NUREG/CR-5512, Volume 1 (NRC, 1992). Another source, although the guidance is more focused on offsite exposures, is NUREG/CR-5453, “Background Information for the Development of a Low-

Level Waste Performance Assessment Methodology," Volumes 1 and 2, issued December 1989 (NRC, 1989).

- Categorize the general types of residual radioactivity at the site (e.g., sediment or soil, deposits in buildings, surface residual radioactivity, surface water, groundwater, or industrial products such as slag).

- Screen out pathways, for each contaminant type, that do not apply to the site.

- Identify the physical processes pertinent to the remaining pathways for the site.

- Separate the list of exposure pathways into unique pairs of exposure media (e.g., source to groundwater, groundwater to surface water). Determine the physical processes that are relevant for each exposure media pair and combine the processes with the pathway links.

- Reassemble exposure pathways for each source type, using the exposure media pairs as building blocks, thus associating all the physical processes identified with the individual pairs with the complete pathway.

The licensee's documentation of the decisions made about inclusion (or exclusion) of the various pathways should be transparent and traceable. An international working group established a methodology for developing models to analyze radionuclide behavior in the biosphere and associated radiological exposure pathways (i.e., the Reference Biospheres Methodology). BIOMOVS II published the methodology in its Technical Report No. 6, "Development of a Reference Biospheres Methodology for Radioactive Waste Disposal," issued September 1996 (SSI, 1996), and included a list of international biosphere features, events, and processes.[6] The report may be useful as a guide for additional information on a logical method to complete the pathway analysis above and include proper justification. Generally, the Reference Biospheres Methodology is more useful for complex sites that may have numerous physical processes that interact in such a way that a number of different exposure groups may need to be investigated to identify the critical group. Additional work has been done on implementing the Reference Biospheres Methodology by a working group of the International Atomic Energy Agency's (IAEA's) Biosphere Modeling and Assessment (BIOMASS) program (IAEA, 1999a, 1999b, 2001). Specifically, IAEA Working Document BIOMASS/T1/WD03, "Guidance on the Definition of Critical and Other Hypothetical Exposed Groups for Solid Radioactive Waste Disposal," may provide additional information on developing a site-specific critical group for situations where the generic critical group is inappropriate (IAEA, 1999b).

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6 Additional features, events, and processes lists are presented and discussed in Appendix C of NUREG-2175, draft "Guidance for Conducting Technical Analyses for 10 CFR Part 61," issued March 2015, that may be appropriate for more complex decommissioning sites (NRC, 2015a). In additional to considering alternative exposure scenarios, for some complex decommissioning sites, the licensee may need to consider central and alternative scenarios, as defined in this volume (see definition of "scenario"). For example, if long-lived residual radioactivity is present at an actively eroding site, alternative scenarios related to future landscape evolution of the site may need to be evaluated to adequately assess the long-term risk associated with residual radioactivity remaining at a site. Consideration of this type of alternative scenario (i.e., an alternative scenario that considers future landscape evolution) differs fundamentally from consideration of alternate ***exposure*** scenarios, which primarily focus on assumptions related to future human behavior and expected land use. In some cases, however, an alternative scenario may cause the exposure pathways to change and require consideration of a new alternative exposure scenario as well (e.g., gully erosion may expose buried residual radioactivity that could expose a recreational user of the site, although the licensee may eliminate a residential scenario, as construction of a residence in the area of active erosion would not be realistic, given its proximity to surface water and uneven topography).

### *I.3.3.3 Guidance on Specific Issues*

*Land Use*

A licensee's assumptions for land use should focus on current practice in the region. The region of concern can be as large as an 80-kilometer (km) (50-mile) radius. To narrow the focus of current land practices, the licensees can use information on how land use has been changing in the region and should give more weight to land use practices either close to the site or in similar physical settings. This can be very important for semirural sites that are being encroached upon by suburban residential development. Reviewers may wish to involve State and local land use planning agencies in discussions, if the licensee has not already requested their involvement.

Potential land uses should be categorized as reasonably foreseeable, less likely but plausible, or implausible. Any land uses that similar properties in the region currently have, or may have in the near future (e.g., approximately 100 years[7]), should be characterized as reasonably foreseeable. Consideration should be given to trends and area land use plans in determining the likelihood of potential land use. The time frame of interest for exposure scenario development could be less than 100 years in certain cases and would depend on such factors as the rate of change in land use patterns in the area, radionuclides of interest, and the time of peak dose. For example, a site with residual cobalt-60, which has approximately a 5-year half-life, would not likely need to explore possible land uses that may exist at the site beyond a few decades, because of the natural decay of the residual material.

Land uses that are plausible, generally because similar land historically was used for the purpose but are counter to the current trends or regional experience should be characterized as less likely but plausible (e.g., rural use of property currently in an urban setting). Implausible land uses are those that, because of generally physical limitations, could not occur (e.g., residential land use for an underwater plot of land).

Land use justifications by licensees often rely on State or local codes, in building or well development to constrain future use. In general, licensees requesting unrestricted release should not rely solely on these factors as reasons to remove pathways or justify the exposure scenario unless (1) the radionuclides have a relatively short half-life (approximately 10 years or less) or (2) the dose from long-lived radionuclides reaches its peak before 100 years. Similarly, licensees requesting unrestricted release should not limit land use exposure scenarios based on commitments or require the enforcement of limitations by the licensee or another party (e.g., a licensee reiterates that the land will remain industrial by stating that the land will not be sold by the licensee after the license is terminated).

Licensees should base justifications of land use on (1) the nature of the land and reasonable predictions based on its physical and geologic characteristics, and (2) societal uses of the land, based on past historical information, current uses of it and similar properties, and what is reasonably foreseeable in the near future. The societal uses of the site in the future should be based on advice from local land planners and other stakeholders on what possible land uses are likely within a time period of around 100 years. The level of justification for the final land uses is inversely proportional to the level of realism assumed by the licensee. Limited

[7] Note that the 100-year timeframe described here is only for estimating future land uses; the licensee must evaluate doses that could occur over the 1,000-year time period specified in the LTR.

justification may be required for bounding analyses, while much more detailed justification, including alternative reasonably foreseeable and less likely but plausible exposure scenario analyses, may be needed for a situation with a smaller degree of conservatism in the analyses.

Additional guidance is available on potential sources of land use information in Appendix M.

## WATERBORNE EXPOSURE PATHWAYS

Removal of waterborne exposure pathways can range from global (e.g., all groundwater pathways) to specific (e.g., no drinking water but agricultural/fish pond use remains). Acceptable justifications are generally based on physical conditions at the site rather than local codes. The licensee should base its justification of water quality and quantity of the saturated zone on the classification systems used by the U.S. Environmental Protection Agency (EPA) or the State, as appropriate. Arguments involving depth to water table, or well production capacity, should have supporting documentation from either the USGS, an appropriate State agency, or an independent consultant.

NRC license reviewers should evaluate the reasons for the classification (e.g., information on water quality used as a basis for eliminating pathways). Where the aquifer is classified as not being a source of drinking water but is adequate for stock watering and irrigation, the licensee can eliminate the drinking water pathway but should still maintain the irrigation and meat/milk pathways. Aquifers may exceed certain constituents and still be able to be used for various purposes, because those constituents may easily be treatable (e.g., turbidity). In cases where the water may be treatable or because the degree of connection between the aquifer and surface water may make the use of the aquifer questionable, the reviewer should involve the EPA or the State, or both, as appropriate, in discussions on reasonable assumptions for the aquifer use.

## AGRICULTURAL PATHWAYS

Agricultural pathways may be removed or modified for reasons such as (1) land use patterns, (2) poor-quality soil, (3) topography, and (4) size of contaminated area. Many justifications may result in modification of the pathways, rather than complete elimination. For example, the poor quality of the soil may make intensive farming activities impractical, but residential gardening may still be reasonable.

Licensees using poor-quality soil as a justification for modifying the agricultural pathways should provide the reviewer with supporting documentation from the Soil Conservation Service, appropriate State or local agency, or an independent consultant. Reviewers should carefully consider whether the state of the soil would reasonably preclude all activities (e.g., because of high salinity of soil) or only certain activities. In most cases, soil quality can reasonably preclude activities such as intensive farming but could allow grazing or small gardens.

When reviewing justifications involving topography, the NRC reviewer should limit speculation of future topographical changes from civil engineering and evaluate the reasonableness of the critical group performing its activities on the current topography, for example, a slope. The licensee should provide supporting documentation in the form of pictures, USGS or similar topographic maps, hand-drawn maps, or a detailed description of how the topography would limit farming. NRC reviewers may wish to visit a site to evaluate the topography firsthand.

## AGE-DEPENDENT CRITICAL GROUPS

The definitions in 10 CFR Part 20 should be used when demonstrating compliance with the requirements of Subpart E. EPA's Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," issued September 1988, should be used when calculating internal exposures by using the intake-to-dose conversion factors, which are based primarily on adults. EPA's "Federal Radiation Protection Draft Guidance for Exposure of the General Public," Volume 59, dated December 23, 1994 indicates that implementing age and sex dependent limits for the general public is difficult and due to uncertainty in the impact of these factors on dose that detailed consideration of age and sex is generally unnecessary.

Since age-based dose conversion factors are not being used, the same dose conversion factors are applied to all individuals. Only in rare exposure scenarios will a nonadult individual receive a higher dose (i.e., intake more radioactive material) than an adult individual in a similar exposure scenario. One example may be related to the milk pathway. Children generally drink more milk annually than adults. If milk were the only pathway that would expose the individual to a dose, then the child would have a slightly higher dose than the adult. But in most situations, especially ones involving multiple pathways, the total intake of the adult is greater than that of a child. Therefore, for most multiple pathway exposure scenarios, such as screening analyses, the average member of the critical group should usually be assumed to be an adult, with the proper habits and characteristics of an adult. As the licensee eliminates pathways or modifies the exposure scenario, the behavior and dietary habits of children may become important. In such cases, the licensees should contact the NRC staff for guidance.

## IMPACT OF AREA ON EXPOSURE PATHWAYS AND SCENARIOS

As discussed above, the default exposure scenario for surface soil assumes large areas of homogeneous surface residual radioactivity. If the area of residual radioactivity is smaller than that needed to support the exposure pathway (e.g., an insufficient area to support the production of the quantity of contaminated crops assumed to be consumed by the receptor), the licensee may propose modifying the exposure pathways to account for the effect area has on the critical group's activities. The licensee can use different approaches to account for smaller areas of elevated activity, including the following two methods:

(1) Reduce the calculated dose by modifying the exposure time or usage parameters accordingly.

(2) Modify the exposure scenario and pathways and/or modify the calculation method to account for the size of the residual radioactivity.

These methods may be built into some dose assessment codes for surficial soil (e.g., RESRAD-ONSITE), but the user should understand how the codes consider the area of contamination in adjusting pathway doses. For example, RESRAD-ONSITE and RESRAD-BUILD are commonly used to perform site-specific dose modeling and to calculate DCGLs. These codes uniquely consider the size of the area of the contaminated zone in calculating the dose for each pathway.

It should be noted, however, that RESRAD-ONSITE and RESRAD BUILD do not adjust the occupancy factors for smaller areas of contamination. As the size of the elevated areas decrease, full occupancy assumptions may become increasingly unrealistic and overly conservative (e.g., by default, RESRAD-ONSITE assumes that the receptor is located in the

center of the contaminated zone). In these cases, the licensee may present arguments for modified occupancy assumptions or alternative exposure scenario assumptions for smaller areas of residual radioactivity, based on the expected habits and characteristics of members of critical groups, as well as the characteristics of the site being evaluated. Certain exposure pathways, such as meat and plant ingestion, are also affected by the size of the elevated area. Therefore, the licensee can present arguments for why the dose from certain pathways may be limited due to the area of the elevated concentration and adjust other parameters, as necessary, to avoid overly conservative dose calculations and associated overly conservative cleanup levels. For example, if the contaminated plant fraction is set to -1, RESRAD-ONSITE adjusts the total amount of contaminated plant products ingested based on the size of the contaminated area. If other dose modeling codes are used, the licensee may justify using lower consumption rates of contaminated homegrown produce or animal products, based on the size of the elevated area.

Table I.4 summarizes the radionuclide-specific factors above the $DCGL_W$ that are allowable for smaller areas of residual radioactivity (i.e., area factors).[8] These factors are (1) provided in lookup tables included in various regulatory documents, (2) calculated using the formula provided for field analysis in the "User's Manual for RESRAD Version 6," and (3) calculated using RESRAD Version 6.5. Differences between area factor values listed for a specific radionuclide can be attributed to a variety of factors, including the use of different models or methods, modeling parameter values (e.g., length parallel to aquifer flow in the RESRAD nondispersion model), and modeling assumptions. The EPA Soil Screening Guidance (EPA, 1996) and RESRAD Field Formula methods are generally more limiting for radionuclides where the dose is attributable primarily from ingestion pathways and comparable to other methods for radionuclides where the dose is attributable primarily to the external exposure pathway.

Some general observations related to cleanup levels for smaller areas and general characteristics of area factors include the following:

- The $DCGL_{EMC}$ value will always be greater than or equal to the $DCGL_W$ value (i.e., area factors will always be larger than 1). In other words, higher concentrations or cleanup levels are allowable for smaller elevated areas compared to larger areas of residual radioactivity to meet the same radiological criteria for license termination.

- With respect to water-independent pathways:

    - RESRAD-calculated area factors for gamma-emitting radionuclides where the risk is dominated by the external exposure pathway generally have more restrictive or lower area factors that vary nonlinearly with the area of contamination (see for example cesium (Cs)-137 and Co-60 in Table I.4).

    - RESRAD-calculated area factors for the inhalation exposure pathway also vary nonlinearly with the size of the contaminated area.

[8] It is important to note that $DCGL_{EMC}$s should be based on dose modeling and not calculated based on area factors reported in this volume or "default" area factors provided in other reference material. Area factors reported in this volume are provided to facilitate discussion of differences in the impact of area on dose for different radionuclides and pathways only. They should not be interpreted as being acceptable for use in developing $DCGL_{EMC}$s.

- RESRAD-calculated area factors for ingestion pathways such as soil ingestion, animal product ingestion, and plant ingestion generally have area factors that scale directly to the size of the contaminated area.

- With respect to water-dependent pathways:

  - RESRAD-calculated area factors for the drinking water and fish ingestion pathways generally have more limiting area factors compared to other water-dependent pathways and can be more limiting than water-independent pathways.

  - RESRAD-calculated area factors for ingestion pathway doses incurred from use of contaminated irrigation water are generally lower and DCGLs are generally higher or less restrictive compared to other water dependent and water-independent pathways, and therefore, area factors for these pathways are generally not limiting and are not as risk significant.

The NRC staff should review the following information provided by the licensee:

- summary table or list of the $DCGL_W$(s) for each radionuclide and impacted medium of concern

- the $DCGL_{EMC}$ for each radionuclide and medium of concern, if Class 1 survey units[9] are present

- the appropriate $DCGL_W$ for the survey method to be used if multiple radionuclides are present

The NRC should review licensee-calculated $DCGL_{EMC}$s to ensure that the values are developed based on dose modeling (i.e., default area factors found in the literature should not be used to assign $DCGL_{EMC}$s). Consideration can be given to site-specific conditions, including the contributions of individual exposure pathways to the overall dose, since exposure scenarios and pathways can vary from one site to another. Additionally, the NRC should carefully review the licensee's approach for considering multiple radionuclides and elevated areas.

[9] Class 1 survey units are impacted areas that are expected to have concentrations of residual radioactivity that exceed the DCGLW.

**Table I.4 Comparison of Area Factor Values from Different References**

| | | 10000 $m^2$ | 1000 $m^2$ | 100 $m^2$ | 10 $m^2$ | 1 $m^2$ |
|---|---|---|---|---|---|---|
| Am-241 | NUREG-1505 | 1 | 1.01 | 1.86 | 13.4 | 109 |
| | NUREG-1575 | 1 | 1.3 | 13.4 | 96.3 | 208.7 |
| | EPA Soil Screening Guidance | 1 | 1.1 | 1.3 | 2.3 | N/A |
| | RESRAD "Field Formula" | | | | 3.2 | 10 |
| | RESRAD-ONSITE, Version 6.5 | 1 | 1.01 | 1.87 | 14.1 | 124 |
| Cs-137 | NUREG-1505 | 1 | 1.1 | 1.41 | 2.41 | 11 |
| | NUREG-1575 | 1 | 1.1 | 1.4 | 2.4 | 11 |
| | EPA Soil Screening Guidance | 1 | 1.1 | 1.3 | 2.6 | N/A |
| | RESRAD "Field Formula" | | | | 3.2 | 10 |
| | RESRAD-ONSITE, Version 6.5 | 1 | 1.14 | 1.41 | 2.41 | 11 |
| Co-60 | NUREG-1505 | 1 | 1.1 | 1.23 | 2.12 | 9.81 |
| | NUREG-1575 | 1 | 1.1 | 1.2 | 2.1 | 9.8 |
| | EPA Soil Screening Guidance | 1 | 1.1 | 1.3 | 2.6 | N/A |
| | RESRAD "Field Formula" | | | | 3.2 | 10 |
| | RESRAD-ONSITE, Version 6.5 | 1 | 1.06 | 1.23 | 2.12 | 9.81 |
| Th-232 | NUREG-1505 | 1 | 1.03 | 1.75 | 3.12 | 12.3 |
| | NUREG-1575 | 1 | 1.1 | 1.8 | 3.2 | 12.5 |
| | EPA Soil Screening Guidance | 1 | 1.1 | 1.3 | 2.3 | N/A |
| | RESRAD "Field Formula" | | | | 3.2 | 10 |
| | RESRAD-ONSITE, Version 6.5 | 1 | 1.05 | 1.82 | 3.31 | 15 |
| U-238 | NUREG-1505 | 1 | 1.04 | 2.27 | 11.1 | 30.5 |
| | NUREG-1575 | 1 | 1.3 | 6.7 | 11.1 | 30.6 |
| | EPA Soil Screening Guidance | 1 | 1.1 | 1.3 | 2.4 | N/A |
| | RESRAD "Field Formula" | | | | 3.2 | 10 |
| | RESRAD-ONSITE, Version 6.5 | 1 | 1.15 | 2.32 | 15.4 | 80.2 |

Notes:

- NUREG-1505 (NRC 1998b), Table 8.1, reports radionuclide-specific area factors calculated using RESRAD-ONSITE, Version 5.7.
- NUREG-1575 (NRC 2000a), Table 5.6, reports radionuclide-specific area factors calculated using RESRAD-ONSITE, Version 5.6.
- Area factors associated with the EPA Soil Screening Guidance (EPA, 1996) are the reciprocals of the area correction factors reported in Table 5.2 of the document. The area correction factors are based on the external exposure pathway only.

As indicated above, when the extent of residual radioactivity becomes smaller, some of the activities are no longer viable as reasonable assumptions for exposure. Generally, the first pathways affected are animal husbandry activities, because of the larger area needed for grazing and growing fodder. As a general rule, as the area gets smaller, the more the exposure scenario transforms into a residential gardener exposure scenario, so long as the initial residual radioactivity begins in the surface soil. For cases where the residual radioactivity is not in the

surficial soil, the original area of residual radioactivity may not be as important in exposure scenario development, because some of the primary transport mechanisms result in redistribution of the radionuclides over larger areas (i.e., groundwater used as irrigation).

One common mistake in licensee submittals is that $DCGL_{EMC}$s are typically not provided for residual radioactivity on building surfaces. When the screening $DCGL_W$ values were published in the *Federal Register* (see Appendix H), associated $DCGL_{EMC}$s were not published. Although newer versions of DandD (Version 2) allow specification of a limited area of contamination, DandD adjusts dose based on area in a simplistic manner. Therefore, the licensee may wish to calculate $DCGL_{EMC}$s for building surfaces using the RESRAD–BUILD computer code.

## OFFSITE EXPOSURE SCENARIOS

As discussed above, in rare situations, the exposure scenario resulting in the highest exposures from the residual radioactivity will be an offsite use exposure scenario. For these evaluations, the dose limits in 10 CFR Part 20, Subpart E, remain applicable, even though the situation may seem similar to the clearance of materials before license termination. In these scenarios, the exposure to the radioactive material will occur, because it has been removed from the current location, and this results in either new or enhanced exposure pathways. For example, a site has poor groundwater characteristics (thereby, allowing the licensee to remove the groundwater pathway from any applicable exposure scenarios), and the reasonably foreseeable land use is either commercial or industrial. The primary contaminant is technetium (Tc)-99, which primarily results in dose through either the groundwater or vegetable pathways, both of which are not applicable to the physical characteristics of the site or land use assumptions. The residual radioactivity is present in the site's topsoil. A possible offsite exposure scenario is where, during construction of any commercial interest on the site after license termination, the removed topsoil is sold for use in a residential setting. In this case, it is likely that the topsoil with residual radioactivity will be unintentionally mixed with other topsoil at the offsite location. Licensees can use generic analyses to screen the importance of offsite uses with such sources as NUREG-1640, "Radiological Assessments for Clearance of Materials from Nuclear Facilities." (NRC 2003b)

Even if offsite use is not considered reasonably foreseeable, offsite exposure scenarios may be less likely but plausible scenarios and should be analyzed as exposure scenarios, to understand the robustness of the analysis.

## DETERMINING THE COMPLIANCE EXPOSURE SCENARIO

In many situations, a licensee will be faced with selecting a compliance exposure scenario from a potentially large suite of exposure scenarios and exposure groups. The licensee is expected to base its demonstration of compliance on the reasonably foreseeable exposure scenario resulting in the highest peak dose during the compliance period, consistent with the definition of the critical group. Licensees may find it advantageous to use an iterative approach to screen all the potential exposure scenarios. This will allow the licensees to focus their more detailed analyses on the important exposure scenarios. Licensees may be able to use information from NUREG/CR-5512 (NRC 1992), NUREG-1640 (NRC 2003b), and NUREG-1717 (NRC 2001), as well as other licensees' analyses to screen their potential exposure scenarios with quantitative methods. Licensees also may be able to provide qualitative arguments to demonstrate that the dose from certain exposure scenarios is bounded by the dose of higher level exposure scenarios (e.g., a residential gardening exposure scenario will bound the dose for the residential

nongardening exposure scenario). The licensees should provide justifications on the basis, method, and results of their exposure scenario screening in their DP.

Even after screening the exposure scenarios, a licensee will likely be left with a few exposure scenarios that may require detailed analyses to determine which will result in the critical group. For licensees with multiple radionuclides, determining the compliance exposure scenario commonly depends on the final mixture of radionuclides. This can provide a dilemma for licensees creating DCGLs. The licensee must show that the final concentrations at the site meet the dose criteria of 10 CFR Part 20, Subpart E. Two possible approaches that the licensee may use to show compliance are, but are not limited to, the following:

(1) Use the most limiting DCGL for each radionuclide, regardless of the exposure scenario, and use the sum of fractions, ignoring the exposure scenario basis for each DCGL. This approach requires limited justification. It will always either estimate the same dose as the individual exposure scenarios or overestimate the dose. Generally, it will greatly overestimate the dose for the individual exposure scenarios.

(2) Commit to demonstrating the final dose for each of the important exposure scenarios in the FSS reports. This approach will require the licensee to establish operational DCGLs to fully use MARSSIM (see Section 2.5).

The licensee needs to provide either a quantitative analysis of, or a qualitative argument discounting the need to analyze, all the exposure scenarios generated from the less likely but plausible land uses. The results of these analyses will be used by the staff to evaluate the degree of sensitivity of dose to overall exposure scenario assumptions (and the associated parameter assumptions). Analyses of less likely but plausible exposure scenarios are not meant to be 'worst-case' analyses and should not use a set of 'worst-case' parameters. Selection of parameters for less likely but plausible exposure scenarios should be consistent with the guidance in this appendix. The reviewer will consider both the magnitude and time of the peak dose from these exposure scenarios. If the peak dose from the less likely but plausible land use exposure scenarios is significant, the licensee would need to provide greater assurance that the exposure scenario is unlikely to occur, especially during the period of peak dose. The licensee may be able to show that the compliance exposure scenario bounds the results of all or many of the exposure scenarios associated with the less likely but plausible land uses.

### I.3.4 Generic Examples

The following examples describe situations where the default pathways may be removed or modified. Note that the examples assume that an adequate level of justification has been provided by the licensee.

#### *I.3.4.1 Removal of Groundwater Pathways*

A licensee has extensive contamination of the upper soil horizon and the upper aquifer, which is unconsolidated, and it wishes to remove the groundwater pathway because the upper aquifer would not be used as a water source. The aquifer shows relatively high levels of microbial activity, turbidity, and nitrates. In addition, adjacent to the site is a small patch of wetlands that shows a great deal of communication with the upper aquifer. The potential yield rate of the upper aquifer is sufficient for domestic use, but there is a better quality, confined aquifer, with a horizon at a depth of approximately 30 meters (100 feet). The licensee has also demonstrated

that the deeper aquifer will not become contaminated from the upper aquifer. Considering all of these reasons in combination, it is questionable whether the upper aquifer would actually be used. Although it may be possible for someone to treat the contaminants and use the aquifer, there are better sources of water easily available. After consultation with the EPA and the State, it is agreed that it would be unreasonable to assume someone would use the upper aquifer as a water source. Therefore, the licensee is allowed to remove the groundwater pathway from the exposure scenario.

*I.3.4.2 Exposure Scenario Development for Buried Residual Radioactivity*

**EXAMPLE 1: SUBSURFACE SOIL**

A site has residual radioactivity buried at a few feet below the surface and the licensee is requesting unrestricted release. The residual radioactivity does not have enough highly energetic gamma emitters to result in an external dose in the current configuration. Two scenarios can be developed (without any other site-specific information): (1) leaching of the radionuclides to the groundwater, which is then used by a residential farmer, and (2) exposure to the buried residual radioactivity by house construction for a resident farmer with the displaced soil, which includes part of the residual radioactivity, spread across the surface. Exposure scenario 2 encompasses all the exposure pathways and, although not all of the source term is in the original position, leaching may occur both from the remaining buried residual radioactivity and the surface soil. In certain cases, alternative intrusion events could be more limiting and should be considered particularly if home construction into the buried residual radioactivity is precluded due to the depth of the residual radioactivity or presence of a cover. Appendix J of this volume describes in greater detail the consideration of residual radioactivity in subsurface soils.

**EXAMPLE 2: EMBEDDED PIPING**

At another site, the licensee is requesting unrestricted release of its site. It is removing the buildings but is evaluating the need to remove the concrete pads, which have embedded piping that contains the residual radioactivity. Two exposure scenarios can be reasonably envisioned. The first involves a resident farmer onsite, who builds a house on the concrete pad, without disturbing the embedded piping. Possible exposure pathways would be external dose from the piping and exposure to leached materials from the piping through groundwater use (e.g., drinking, irrigation). The second exposure scenario is similar to the building renovation exposure scenario discussed in Example 1, where the concrete pad and piping are removed from the site during a future construction project. The licensee should investigate both exposure scenarios to find the limiting scenario.

*I.3.4.3 Exposure Scenario Development for Restricted Release*

For this example, the site restrictions planned for an alternate site include a restriction on the deed that the property can be used only for parkland, and an engineered cover is placed over the residual radioactivity. The engineered cover is contoured for use as parkland and has a vegetative cover (i.e., not a mound covered in rip-rap). Three exposure scenarios are easily envisioned for the restricted release analysis. The first is recreational use of the property as a city park or golf course, which would limit exposure scenarios to possible external exposure. The second would involve offsite use of groundwater that contains radionuclides leached from the buried residual radioactivity. The default offsite user would be a resident farmer using the

groundwater for all water needs. The third exposure scenario would be a worker maintaining the park.

The doses assuming the loss of institutional control (i.e., the deed restriction) immediately following license termination (or time=0 years) and degradation of the engineered cover also must be evaluated. Scenarios should consider how critical groups could be exposed to the residual radioactivity through disruptive events.

Consider a residential farmer who uses groundwater from the aquifer located under the site. An engineered cover may become compromised by the placement of buildings. The cover may still perform in some degraded function for some period of time. Whether buried residual radioactivity is transported to the surface by the construction of a basement under the resident farmer's house would depend on the thickness of the engineered cover. If typical basement depths are deeper than the engineered cover's thickness, some portion of residual radioactivity would be transported to the surface, mixed with the "clean" cover material, and spread over the site. If the typical basement depth is shallower than the engineered cover thickness, other disruptive events such as well construction or large-scale excavation of material may need to be considered to evaluate a scenario where residual radioactivity is brought to the surface (see Appendix J).

In another scenario, the engineered cover may become degraded from erosion and residual radioactivity redistributed through hydrological processes. The reasonableness of this scenario would depend on the thickness and erosion-resistance of the engineered cover (see Section 3.5 for additional information on performance assessments for engineered barriers).

## I.4 Criteria to Establish Conceptual and Mathematical Models

### I.4.1 Introduction

Analyzing the release and migration of radionuclides through the natural environment and engineered systems, at a specific site, requires the licensee to interpret the nature and features of the site so that the site can be represented by mathematical equations (i.e., mathematical models). This simplified representation of the site is commonly referred to as the conceptual model of the site.

Figure I.3 depicts the process of conceptual model development. In dose assessments, developing a conceptual model involves making an abstraction of site data into a form that is capable of being modeled. This development should generally involve making simplifying assumptions, including simplification of the appropriate governing equations, to reflect the physical setting. These simplifying assumptions are usually made in describing the geometry of the system, the spatial and temporal variability of parameters, the isotropy of the system, and the influence of the surrounding environment. The conceptual model should provide an illustration or description of site conditions, to show, or explain, contaminant distributions, release mechanisms, exposure pathways and migration routes, and potential receptors. In other words, the conceptual model should explain or illustrate how radionuclides enter, move through and/or are retained in, and leave the environment.

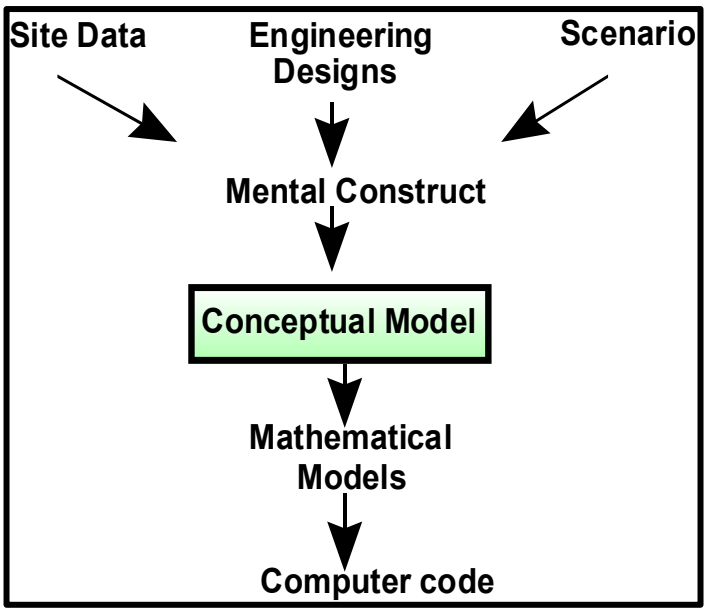


**Figure I.3 Conceptual Model Development**

As shown in Figure I.4, developing a conceptual model at a site is Step 3 of the decommissioning decision framework (see Figure 1.2 of this volume). Conceptual model development follows after assimilation of site data (Step 1) and definition of scenarios (Step 2), because information from these two steps feeds into its development. In other words, the conceptual model should be based on what is known about the site from data and information gathered as part of Step 1, and how the site evolves during the period covered by the analysis, based on the assumed land use defined under Step 2.

Mathematical models are a quantitative representation of the conceptual model. Because the conceptual model provides the linkage between site conditions and features (Steps 1 and 2) and the computer code(s) (with its associated mathematical models) used in the dose analysis (Step 4 of the decommissioning framework), it is a key step in a dose assessment and should not be taken lightly.

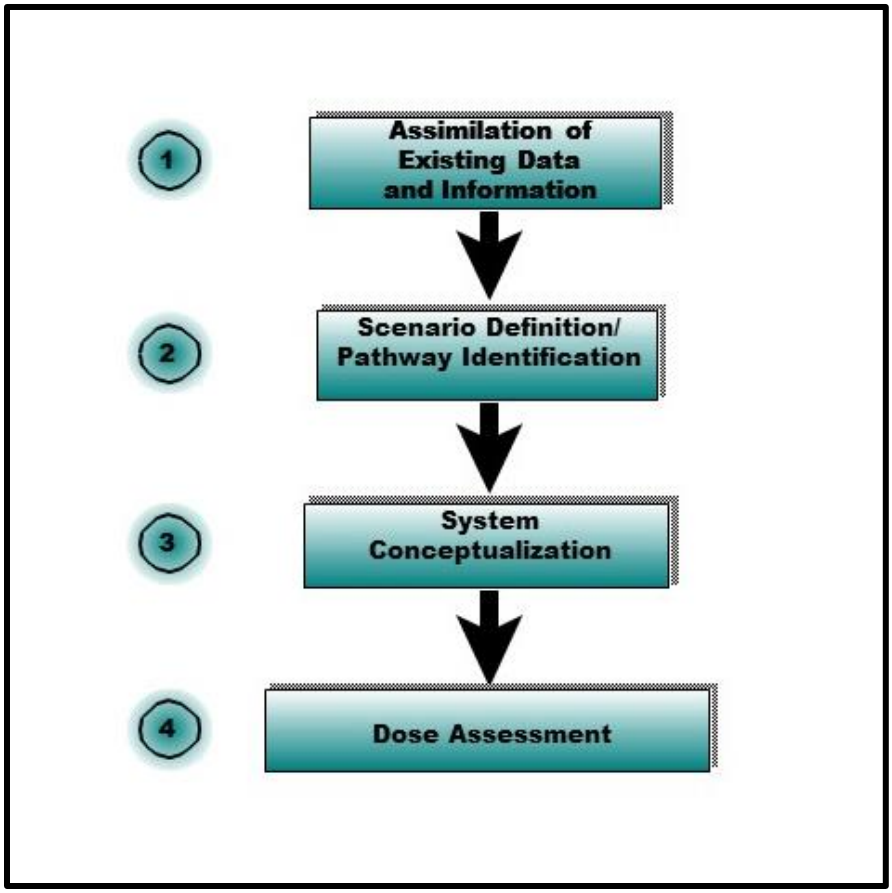


**Figure I. 4 Decommissioning Decision Framework**

### I.4.2 Technical Issues

Uncertainties in conceptual models can be large and possibly even larger than uncertainties in parameters used in the analysis (James and Oldenburg, "Linear and Monte Carlo Uncertainty Analysis for Subsurface Contaminant Transport Simulation," issued November 1997). Thus, conceptual model uncertainties can be a significant source of uncertainty in the overall dose assessment. Uncertainties in the conceptual model(s) are generally caused by incomplete knowledge about the natural system being analyzed and differing views about how to interpret data representing the system.

Development of conceptual models is a subjective process based on the interpretation of limited (or in most cases, sparse) site data. From these limited data, the licensee should determine the key processes and features at the site and how they are likely to affect the movement of radionuclides through the environment. Because the conceptual model of the site is based on incomplete information, it is possible that multiple interpretations of the same data can be derived. A licensee should also determine the appropriate level of simplification acceptable for representing the site. An overly simplified conceptual model may leave out key site features or conditions that are important in estimating where radionuclides are likely to be transported (thus, where people might be exposed) and when they might get there (thus, the radionuclide concentration when it arrives). On the other hand, an overly complex conceptual model may introduce unnecessary uncertainty and costs into the analyses. As a broad example, simple

models contained in screening codes may oversimplify features and processes at a specific site. The licensee also should ensure that the model provides the appropriate level of detail. It is important that the conceptual model have sufficient detail and scope for a license reviewer to be able to assess the appropriateness of the computer codes used in the analysis and the defensibility of the assumptions made. In summary, key issues in developing and presenting the conceptual model are (1) identifying the important site features and processes that need to be included in the conceptual model, (2) deciding among possible competing interpretations of the site data, and (3) determining the level of detail needed to describe those features and processes.

Some important staff insights gained from reviewing decommissioning dose assessments and performance assessments are summarized below. Insights are grouped into those related to (1) model abstraction and (2) model simplification. Issues associated with model abstraction and simplification are generally important for more complex sites, while screening models and codes discussed in the next section I.4.3 may be appropriate for simpler sites provided the underlying assumptions of the conceptual models for those models and codes are consistent with site conditions.

#### *I.4.2.1 Model Abstraction*

Some level of abstraction is required to translate the concepts of a conceptual model into mathematical terms. An abstracted model can be something as simple as a data value or lookup table. The methods to produce abstracted models are often not particularly rigorous and can introduce quantified uncertainties and biases. Important aspects can be lost in the abstraction process, and, for this reason, the reduction is often undertaken in such a way that it produces a conservative result. It is therefore important to clearly document the model abstraction and recognize the potential impacts due to the abstraction.

Insights gained on abstracting models of hydrogeological systems include the following:

- Very complicated sites may not need to have all geological and hydrogeological features and processes represented in the model, so that model abstraction may be less problematic than originally estimated.

- Many of the processes that govern transport of radionuclides in the unsaturated zone are essentially the same as those that govern transport in the saturated zone. However, effective hydraulic properties have a nonlinear dependence on soil moisture content, so that flow in the unsaturated zone can be strongly influenced by extreme, but not necessarily uncommon, conditions.

Additional important insights gained from the NRC review of examples of model abstraction include the following:

- Code selection for a particular modeling exercise should be judicious to ensure that code limitations do not lead to nonconservative or unrealistic dose modeling predictions (e.g., lack of consideration of complex source terms, important transport processes such as diffusion and dispersion, or complex flow systems).

- Model abstractions can successfully represent the essential elements of the system being simulated, increase computational efficiency, enable a more complete evaluation

of model and parameter uncertainty, and lead to a better understanding of the system being simulated.

- Additional complexity can and should be added to a model if certain processes or parameters are found to be important (i.e., if the additional complexity significantly influences the results).

- Important pathways of exposure and scenarios should be evaluated to ensure that the most limiting pathways and scenarios are considered. In some cases, it is not intuitive to determine the most risk-significant exposure scenario.

- Source term assumptions may have a significant impact on modeling results (e.g., source orientation and geometry, source distribution, and elevation of release).

### *I.4.2.2 Model Simplification*

Model simplification is the process for reducing the complexity of a numerical model into a simpler numerical model while still maintaining the validity of the simulation results. NUREG/CR-6884, "Model Abstraction Techniques for Soil-Water Flow and Transport," issued December 2006 (NRC 2006), presented a systematic and objective approach to model simplification relevant to subsurface flow and transport modeling. This approach included (1) justifying the need for the model simplification, (2) reviewing the context of the modeling problem, (3) selecting applicable model simplification techniques, (4) determining model simplification directions, and (5) simplifying the complex model in each direction. When performing a model simplification, various categories of techniques are relevant to the subsurface flow and transport modeling. These categories include selecting from a predefined hierarchy of models, changes in spatial dimensionality (e.g., three dimensional to two dimensional), and scale change, including upscaling, aggregation, and metamodeling.

The model simplification process starts with an existing complex model that can be calibrated and used in simulations. Justifying the need for the model simplification is usually associated with unsatisfactory results from the complex model in some way. This can either be related to the complex model being too expensive, too large, and too difficult to run and calibrate; input is too hard to obtain; or output is difficult to understand. After the model simplification process, the resulting model output should provide information that is both necessary and sufficient to make a decision on the issue(s) of interest.

Some important insights gained from the NRC review of previous examples of model simplifications include the following:

- While in many cases, simpler models and codes developed to demonstrate compliance with regulatory criteria are purposefully constructed to err on the side of higher dose, the use of simpler models and codes does not guarantee conservative results. Irrespective of the level of complexity of the model used to facilitate decision-making, the analyst should ensure that the model is adequately supported and that the impact of model simplification is well understood.

- The necessary level of complexity of a model is dependent on many factors, including the complexity of the site, uncertainty in site parameters and processes, the questions that are being addressed by the models, and the safety margin.

- With respect to physical dispersion, models that consider dispersion can generally provide more realistic estimates of concentrations and dose compared to simpler models that do not consider dispersion. For example, more complex three-dimensional groundwater flow and transport models can explicitly consider longitudinal and transverse dispersion, while one-dimensional models may consider no dispersion at all or longitudinal dispersion, or only implicitly consider dispersion. Consideration of physical dispersion can lead to higher or lower concentrations (and the resultant dose), or to no significant impact at all, depending on such factors as the radionuclides driving the results, the presence of multiple sources, receptor locations, numerical model construction decisions, and site complexity (e.g., heterogeneity). Because, in many cases, dispersion assumptions can significantly affect the results, the treatment of dispersion in a model should be adequately studied to ensure that concentrations and dose are not underestimated.

- The essential elements of a complex, three-dimensional groundwater flow and transport model can be identified and considered in a simpler flow and transport model. However, data of sufficient quality needed to calibrate the complex model must be available and a sufficient number of intermediate outputs and observations extracted to ensure that the simpler model accounts for risk-significant processes. If this approach is used, the model abstraction can sometimes be helpful in identifying and communicating key parameters and processes that are difficult to identify or explain using the complex model alone.

For complex sites, when relatively simple models are used to demonstrate compliance with regulatory criteria, the analyst should provide sufficient information (e.g., relatively more and less complex model comparisons) to show that the model simplifications do not lead to a significant loss of fidelity in the results that could be important to decision-making. Appendix F contains additional information about conceptual site model development, model abstraction and simplification pertaining to hydrological models.

### I.4.3 Recommended Approach

#### *I.4.3.1 Screening*

An acceptable dose assessment analysis need not incorporate all the physical, chemical, and biological processes at the site. The scope of the analysis, and accordingly, the level of sophistication of the conceptual model, should be based on the overall objective of the analysis. A performance assessment conceptual model can be simple, if it still provides satisfactory confidence in site performance. For an initial screening analysis, little may be known about the site from which to develop a conceptual model. Computer codes used for screening analyses are generally intended to provide a generic and conservative representation of processes and conditions expected for a wide array of sites. Accordingly, the generic conceptual model in such codes may not provide a close representation of conditions and processes at a specific site. Such a generic representation is still acceptable, as long as it provides a conservative assessment of the performance of the site.

In general, the conceptual models within DandD are expected to provide a conservative representation of site features and conditions. Therefore, for screening analyses, the NRC staff should consider such generic conceptual models to be acceptable, provided it is acceptable to assume that the initial radioactivity is contained in the top layer (building surface or soil) and the remainder of the unsaturated zone and groundwater are initially free of residual radioactivity. In

using DandD for site-specific analyses, it is important to ensure that a more realistic representation of the site that is consistent with what is known about the site would not lead to higher doses. Table I.5 lists some site features and conditions that may be incompatible with the generic conceptual models within DandD. The relative importance of the incompatibilities varies with the exposure scenario and radionuclides involved. More information on the assumptions of the model is available in the development documentation (e.g., the NUREG/CR-5512 series).

For any site where it is known that one or more of these conditions or features are present, the licensee should provide an appropriate rationale on why the use of the DandD should not result in an underestimation of potential doses at the specific site.

As an example, DandD is inappropriate for analyzing sites that contain hydrogen H-3 and C-14 in soil, because DandD considers only the inhalation dose from particulates in the air and does not consider the loss of H-3 and C-14 from the soil to the air as a gas or vapor. To adjust results from the DandD resident farmer exposure scenario for analyzing sites that contain either H-3 or C-14 (Haaker, “Upper bound for inhalation dose from carbon-14 vapor and tritium vapor,” published June 1999) (Haaker, 1999), (1) determine the area of the contaminated zone, (2) run DandD for the site with only H-3 or C-14, (3) read the associated activity ratio factor for the given area from Figure I.5, and (4) estimate the potential missed dose by multiplying the inhalation dose calculated from DandD by the activity ratio factor to account for the dose associated with the gas or vapor phase.

**Table I.5 Site Features and Conditions that may be Incompatible with those Assumed in DandD**

- sites with highly heterogeneous radioactivity
- sites with wastes other than soils (e.g., slags and equipment)
- sites that have multiple source areas
- sites that have contaminated zones thicker than 15 cm (6 in.)
- sites with chemicals or a chemical environment that could facilitate radionuclide releases (e.g., colloids)
- sites with soils that have preferential flow conditions that could lead to enhanced infiltration
- sites with a perched water table, surface ponding, or no unsaturated zone
- sites where the groundwater discharges to springs or surface seeps
- sites with existing groundwater contamination
- sites where the potential groundwater use is not expected to be located immediately below the contaminated zone
- sites with significant transient flow conditions
- sites with significant heterogeneity in subsurface properties
- sites with fractured or karst formations
- sites where the groundwater dilution would be less than 2,000 $m^3$ (70,000 $ft^3$)
- sites where the overland transport of contaminants is of potential concern
- sites with radionuclides in soil that may generate gases (i.e., H-3 or C-14)
- sites with stacks or other features that could transport radionuclides to result in a higher concentration offsite than onsite

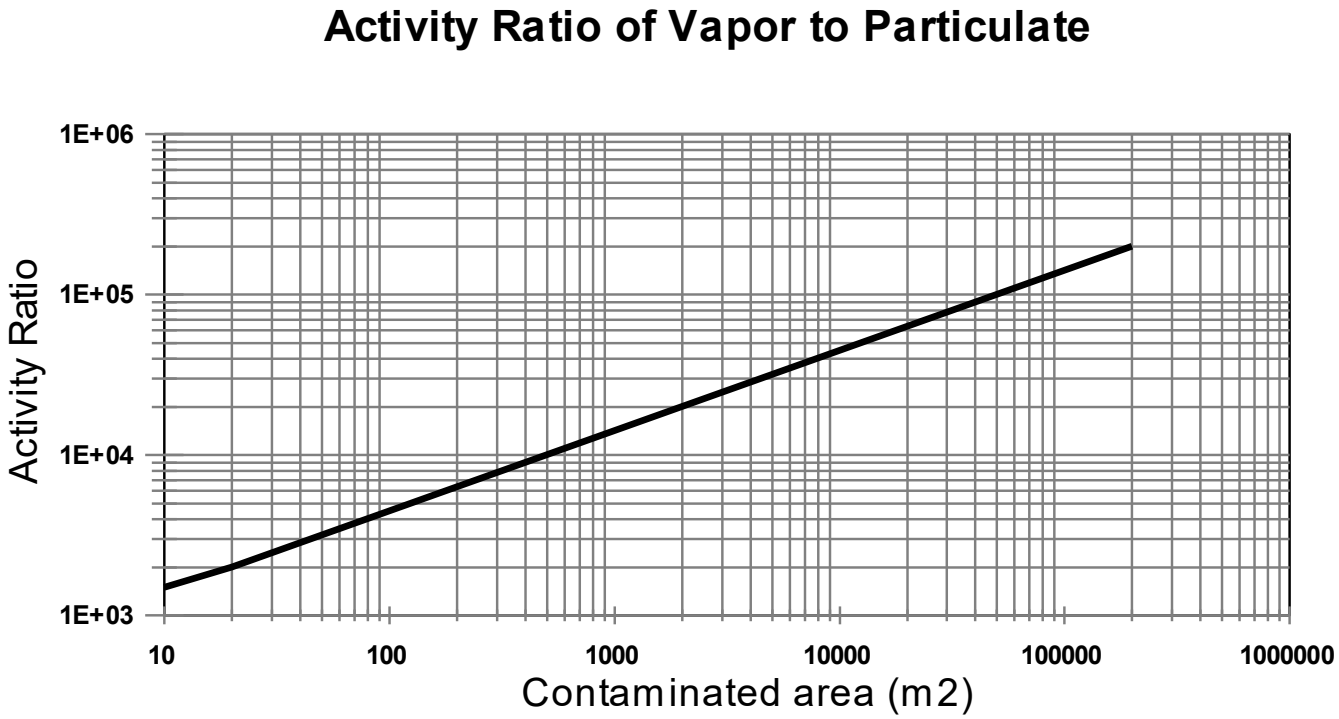


**Figure I.5 Activity Ratio of Vapor to Particulate as a Function of Contaminated Area (Used to Account for Missed H-3 and C-14 Inhalation Dose from Vapor Phase in DandD Residential Exposure Scenario)**

*I.4.3.2 Site-Specific Analyses*

For site-specific analyses, the intent is to provide a more realistic assessment of doses based on more site-specific information and data. Presumably for such analyses, more is known about the site from which to develop a conceptual model. For site-specific analyses, the licensee should provide a schematic or verbal description of the problem that it is attempting to analyze. Even when using a computer code that has a predefined conceptual model, it is important for the licensee to identify any site features or conditions that may differ from those assumed in the code. In developing a site-specific conceptual model or identifying potential limitations with a predefined conceptual model, the licensee should consider the issues listed in Table I.6.

Because conceptual models are developed based on limited data, in most cases, more than one possible interpretation of the site can be justified based on the existing data. The licensee should address this uncertainty by developing multiple alternative conceptual models and proceeding with the conceptual model(s) that provides the most conservative estimate of the dose and yet is consistent with the available data. Consideration of unrealistic and highly speculative conceptual models should be avoided. Consistent with the overall dose modeling framework of starting with simple analyses and progressing to more complex modeling, as

warranted, it may be advisable for the analyst to begin with a simple, conservative analysis that incorporates the key site features and processes and progress to more complexity only as merited by site data. It is important to stress that a simple representation of the site, in itself, does not mean that the analysis is conservative. It is incumbent on the licensee to demonstrate that its simplification is justified, based on what is known about the site and the likelihood that alternative representations of the site would not lead to higher calculated doses.

**Table I.6 Issues to be Considered in Developing a Site-Specific Conceptual Model**

- whether a more realistic representation of the site would lead to higher doses
- whether the conceptual model accounts for the most important physical, chemical, and biological processes at the site
- whether the conceptual model adequately represents responses to changes in stresses
- whether the conceptual model includes consistent and defensible assumptions

In general, there are two primary areas of the dose analysis where the conceptual model is expected to change from one site to another; these are related to the source term and environmental transport. Aspects of the analysis related to the exposure pathways in the biosphere and dosimetry are largely determined by the exposure scenario and the assumed behavior of the critical group. Accordingly, models related to the exposure pathways in the biosphere and dosimetry should not change from one site to another, unless there is a significant change in the exposure scenario and associated critical group. The principal environmental transport pathways that should have to be considered in a dose assessment are groundwater (including transport through the unsaturated zone), surface water, and air.

The conceptual model of the source area should describe the contaminants and how they are likely to be released into the environment. Specifically, it should describe key features and processes such as the infiltration of water into the source area, the geometry of the source zone, the distribution of contaminants, release mechanisms, the physical form of the contaminants, near-field transport processes, and containment failure. If the contaminants are assumed to be uniformly distributed, this is an important assumption that needs to be justified because, in general, contaminants may not be uniformly distributed (see discussion under Section I.2 of this appendix). The source description should clearly identify how the contaminants are assumed to be released from the media. Common release mechanisms are diffusion, dissolution, surface release, and gas generation. The source description should also identify key processes and features that may retain or limit the release of contaminants from the source area (e.g., solubility and sorption). In addition, the description of near-field transport should state assumptions made about the dimensionality. In general, the assumption of one-dimensional vertical flow should be appropriate, unless there is some type of barrier present that may hinder flow in the vertical direction. The description of the source term should also

describe failure mechanisms for any containment (e.g., corrosion, concrete degradation, or cover degradation), if containers or other forms of containment are present.

With respect to source term models, release mechanisms or models include simple wash off, solubility-limited release due to waste form dissolution, and diffusion-controlled release.[10] Release rates are typically expressed as a fractional release rate. Dissolution/precipitation is more likely to be the key process in locations such as at a point source, an area where high contaminant concentrations exist, or where steep pH or oxidation-reduction (redox) gradients exist. Adsorption/desorption will likely be the key process controlling inorganic contaminant migration in areas where the naturally present constituents are already in equilibrium and only the anthropogenic constituents (contaminants) are out of equilibrium, such as in areas far from the point source (EPA, 1999a; and EPA, 1999b).

Use of a $K_d$ model results in the release of residual radioactivity decreasing over time as the source is depleted. Figure I.6 illustrates the depletion of a uranium source for two different release models: (1) solubility-limited release at a value of $3.5 \times 10^{-05}$ moles per liter (mol/L), and (2) desorption from a solid for six different assumed $K_d$s (0.1, 1, 10, 50, 100, 200, 500, and 1,000 L/kilogram (kg)). The total source is depleted more rapidly when solubility control is assumed (at a solubility of $3.5 \times 10^{-05}$ mol/L) compared to a leaching model using a $K_d$ greater than around 50 L/kg, although the release rates may be higher initially for the $K_d$ model. At lower $K_d$s (i.e., less than about 50 L/kg), Figure I.6 shows the sorption model would deplete the total source more quickly than the solubility model. The $K_d$ below which source depletion is more rapid than solubility limited release is a function of the bulk soil density, effective porosity, assumed solubility limit, and source inventory. Important characteristics of the curves shown in Figure I.6 are the following:

- Release rates (i.e., mass released per unit time) in a solubility-limited model are constant over time and the source decay curve is linear.
- The fractional release rate (i.e., fraction of source released per unit time) in a $K_d$ model is constant over time (i.e., quantity released decreases over time) and the release curve is similar to an exponential, first order decay curve.

[10] DUST-MS (Sullivan, 1996) has several models available to characterize the source term or rate of release of residual radioactivity from the source to infiltrating groundwater. The release rate depends upon the physical and chemical form of the radionuclides in the wastes. DUST-MS has four process models including, rinse with partitioning, diffusion, uniform degradation, and solubility-limited release. RESRAD-OFFSITE 4.0 also offers more complex source term models compared to RESRAD-ONSITE including many of the models in DUST-MS.

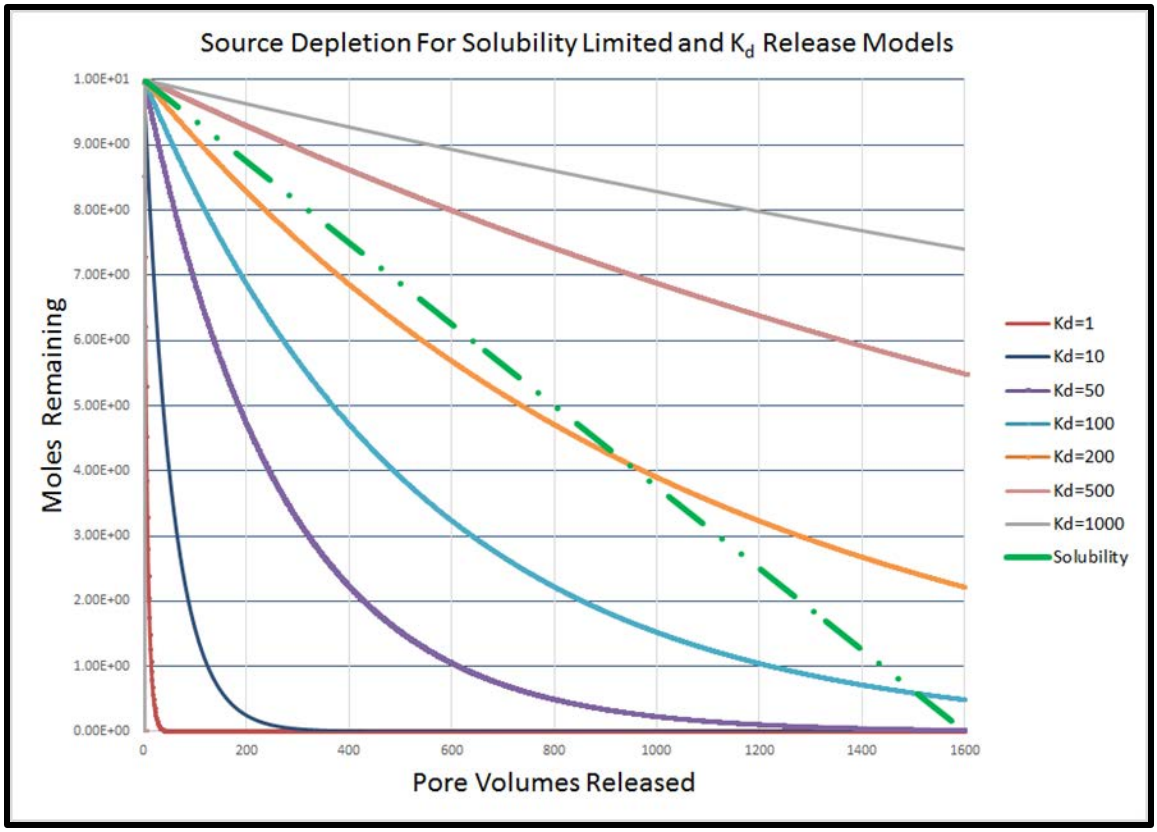


**Figure I.6 Source Depletion for Solubility and $K_d$ Release Models (for Six Values of $K_d$ Expressed in L/kg)**

The conceptual model of the groundwater pathway should describe how contaminants could migrate through the unsaturated and saturated zones to potential receptors (e.g., a well, spring, or surface water). Essential features that should be included in the conceptual model include hydrostratigraphic units; boundary conditions; the physical and chemical form of the residual radioactivity (e.g., dissolved, suspended sediment, gas, speciation, complexation), the structural features of the geology (i.e., those that influence contaminant transport such as fractures, faults, and intrusions), and geochemical conditions and gradients important to contaminant transport. Important processes that should be characterized include the dimensions and state conditions (e.g., steady state) of flow, dimensions and state conditions of transport (e.g., dispersion), chemical and mass transfer processes (e.g., sorption, precipitation, complexation), and transformation processes (e.g., radioactive ingrowth and decay). Although contaminant migration through both the unsaturated and saturated zones is best represented in three dimensions, it may be appropriate to assume only one or two dimensions, if this provides a more conservative representation of contaminant migration, or if it can be demonstrated that migration in one or more other directions is not expected to result in exposure to potential receptors.

The conceptual model of the surface water pathway should describe potential contaminant migration to potential receptors through surface water bodies, such as lakes, streams, channels, or ponds. Essential features that should be included in the conceptual model include: the geometry of the surface water body (i.e., boundaries and boundary conditions), the physical

form of the contaminants (e.g., dissolved or solid), and physical and chemical properties. Key processes that should be described include: the dimensions and state conditions of flow and transport, chemical and mass transfer processes (e.g., sorption, precipitation, volatilization), and transformation. One key boundary condition that should be described is how the contaminants are expected to initially mix or interact with the surface water.

The conceptual model of the air pathway should describe potential contaminant migration through the air to potential receptors. Essential features that should be included in the conceptual model are similar to those for the other environmental pathways—namely, the geometry (i.e., boundaries and boundary conditions), form of contaminants (e.g., particulates or gases), and physical and chemical properties. Key processes that should be described include the dimensions and state conditions of flow and transport and the transformation processes.

## SITE-SPECIFIC COMPUTER CODES

Three common computer codes used for site-specific analyses are RESRAD-ONSITE, RESRAD-OFFSITE, and RESRAD–BUILD[11]. All have predefined conceptual models. Therefore, in using these codes, it is important for the licensee to demonstrate that key site features and conditions are consistent with the modeling assumptions within the codes or, where they are not consistent, the analysis may not result in an underestimation of potential doses. Additional information is provided in Section I.5.3 regarding the built-in conceptual models in these codes.

## LIMITATIONS OF SITE-SPECIFIC COMPUTER CODES

In general, the conceptual models within the RESRAD family of codes are expected to provide an acceptable generic representation of site features and conditions. Table I.7 lists some specific site features and conditions that may be incompatible with this generic representation. At any site where it is known that one or more of these conditions or features are present, the licensee should provide appropriate justification for use of the computer code.

[11] The RESRAD family of codes are commonly used by decommissioning licensees to perform dose modeling to support license termination under 10 CFR Part 20, Subpart E. It is important to note that most of the information on conceptual models in this section pertains to RESRAD-ONSITE and RESRAD-BUILD. While less information is provided about RESRAD-OFFSITE, RESRAD-OFFSITE typically has the same functionality as RESRAD-ONSITE and much more.

**Table I.7 Site Features and Conditions that may be Incompatible with the Assumptions Made in RESRAD-ONSITE**

- sites with highly heterogeneous radioactivity
- sites with wastes other than soils (e.g., slags and equipment)
- sites with multiple source areas
- sites that have chemicals or a chemical environment that could facilitate radionuclide releases
- sites with soils that have preferential flow conditions that could lead to enhanced infiltratio
- sites where the groundwater discharges to springs or surface seeps
- sites where the potential groundwater use is not expected to be located in the immediate vicinity of the contaminated zone
- sites with significant transient flow conditions
- sites with significant heterogeneity in subsurface properties
- sites with fractured or karst formations
- sites where overland transport of contaminants is of potential concern
- sites with stacks or other features that could transport radionuclides off the site at a higher concentration than on site

## I.4.4 Generic Examples

### *I.4.4.1 Screening*

A hypothetical research and development facility is authorized to use radiological chemicals through an NRC license. Because the research and development facility plans to discontinue its use of radioactive material, it wants to decommission the facility and terminate its license. An HSA reveals that the use of radioactive material was limited to a single building within the facility. The floor area of the facility is estimated to be 560 $m^2$ (6,000 $ft^2$). The wall area is 430 $m^2$ (4,600 $ft^2$). In addition, an outside area of roughly 930 $m^2$ (10,000 $ft^2$) was used for dry storage of chemicals. A preliminary characterization program has determined that approximately 10 percent of the building floor area and 5 percent of the wall area are contaminated with Cs-137 and Co-60. Surficial soils covering an area of approximately 2,500 $m^2$ (27,000 $ft^2$) are contaminated from windblown dust and runoff from spills in the storage area. The soils are also contaminated with Cs-137 and Co-60.

The licensee proposes to use a screening analysis, using DandD, to demonstrate compliance with the LTR. A building occupancy exposure scenario is assumed for the building and a residential farmer exposure scenario is assumed for the contaminated soils. Based on what is known about the site, the licensee certifies that the use of the generic conceptual models within DandD is appropriate for the analysis.

*I.4.4.2 Site-Specific*

A hypothetical manufacturing facility has a former radioactive waste burial area that may be decommissioned for unrestricted release. Radioactively contaminated trash was previously buried in 0.2-$m^3$ (55-gallon) drums, in trenches covering an area of roughly 2,000 $m^2$ (22,000 $ft^2$). The trenches, which are roughly 0.9 m (3 ft) deep, are covered with 1.2 m (4 ft) of native soil. A review of site operating records shows that the radionuclides of concern are natural uranium, enriched uranium, and natural thorium.

Based on information from the local county agricultural extension office and published reports, the geology and hydrogeology at the site are described as follows. This description shows that no site features or conditions in Table I.7 are present at this site.

> The surface geology at the site contains 14 to 27 m (46 to 89 ft) of till consisting primarily of fine, silty sand to sandy silt with narrow, discontinuous sand lenses. Sandstone bedrock underlies the unconsolidated till. A shallow unconfined aquifer occurs in the unconsolidated till. The average depth to the water table ranges between 3 and 4 m below the land surface. The mean horizontal hydraulic conductivity is roughly 60 m/y (197 ft/y). The average vertical hydraulic conductivity of the till is estimated to be an order of magnitude less. The hydraulic gradient is estimated to range between 0.006 and 0.021. The mean precipitation at the site is roughly 0.8 m/y (30 in./y). The site is located in the reach of a surface water drainage basin that has a drainage area of approximately 500,000 $m^2$ (5.4 million $ft^2$).

The licensee assumes a residential farmer exposure scenario as a reasonable future land use and proposes to use the RESRAD-ONSITE computer code for the dose analysis. Because the contaminated media is trash, an assumption is made that the trash degrades and becomes indistinguishable from soil. In addition, the metal drums are assumed to have degraded away. Given the relative short lifespan for metal drums and the long half-life of the radionuclides, this should be a reasonable assumption. The cover is also assumed to be breached through the construction of a basement for the house. The contaminated soil is assumed to be uniformly mixed with the excavated cover. Because the trash is assumed to be indistinguishable from soil, it is also assumed that, once the cover is breached, the future hypothetical farmer may not recognize the contaminated material as contaminated. The licensee also assumes that the hypothetical future well is located at the center of the residual radioactivity because of limited bases for assuming otherwise.

The licensee determines that the other aspects of conceptual models within RESRAD-ONSITE are acceptable for analyzing the problem.

## I.5 Criteria for Selecting Computer Codes/Models

### I.5.1 Introduction

Dose assessment commonly involves the execution of numerical model(s) that mathematically represent the conceptual model of the contaminated site. The numerical models used to implement the mathematical equations are usually linked via the conceptual model and codified in a software package known as "the code." The words "code" and "model" are frequently used to express the software package, including the embedded numerical models or the specific models contained in the code. For example, "DandD code" may refer to the software package, including the associated exposure models (e.g., the water use model, food-ingestion pathway model, inhalation-exposure model) embedded in the code. The "DandD model" may also refer to DandD software, the DandD conceptual model, or any of the numerical models, or the group of models used in the code (e.g., DandD groundwater model). Within the context of this volume, the word "code" will refer to the software package and the associated numerical models. However, the word "model" will refer to the mathematical representation of the conceptual model, including representation of the specific exposure scenario and pathways. This section describes the process and criteria used in the selection of codes and models for dose assessments.

The codes and models used in the dose assessment can be either generic screening codes/models or site-specific codes/models. Regardless of the intent of the use of the code/model (e.g., for screening or site-specific analysis), the reviewer should ensure that the licensee properly documents and verifies the dose assessment codes/models and the associated databases in accordance with rigorous QA/QC criteria acceptable to the NRC. Currently, the only acceptable generic screening code is DandD. If the licensee uses site-specific models/codes, it should justify the conceptual model used (see Section I.4.3.2 of this appendix). The NRC staff should also review the source term model(s), the transport models, the exposure models, and the overall dose models and assess the QA/QC documentation and level of conservatism of any alternative code/model.

This section describes the generic issues associated with the selection of the screening and site-specific codes/models that the NRC staff may encounter and recommends approaches and criteria for its acceptance of the codes/models. In addition, this section presents a generic description of common dose assessment codes, DandD, RESRAD-ONSITE and RESRAD–BUILD. The NRC developed or funded (in part) the development of these codes. In addition, the NRC staff and licensees have used these codes to demonstrate compliance with the dose criteria in Subpart E.

### I.5.2 Issues in Selection of Computer Codes/Models

The major issues associated with the selection of computer codes/models include the following:

- **Generic criteria for the selection of computer codes/models:** This issue pertains to the NRC staff's review criteria of code aspects related to QA/QC requirements, specifications, testing, verification, documentation, interfacing, and other features related to uncertainty treatment approaches.

- **Acceptance criteria for the selection of site-specific codes/models:** This issue pertains to the NRC staff's review of additional specific requirements for justifying the

use of the conceptual model, the numerical mathematical models, the source term model and its abstraction, and the transport and exposure pathway models.

- **Options for the selection of deterministic or probabilistic site-specific codes:** This issue pertains to the NRC staff's review of the justification to support the decision to use either of these two approaches.

A generic description of the DandD Version 2 is presented below to familiarize users with this code. Further, the rationale for the development of DandD Version 2 and the issue of excessive conservatism in DandD Version 1 are addressed. A description of the approaches to minimize such excessive conservatism, using DandD Version 2, site-specific input data, or use of other models/codes is included.

For a site-specific analysis, the NRC staff should accept any model or code that meets the criteria described below in "Generic Criteria for the Selection of Codes/Models." However, the staff is expected to conduct a more detailed and thorough review of less common codes/models (e.g., codes other than DandD and the RESRAD family of codes), specifically those developed by licensees. The NRC sponsored development of the probabilistic RESRAD-ONSITE (Version 6) and RESRAD–BUILD (Version 3) codes for site-specific analysis[12]. These have already been reviewed for QA/QC and are acceptable.

The selection of appropriate models/codes for complex sites may also present challenges. For example, sites with multiple source terms, with significant groundwater or surface water contamination, or sites with existing offsite releases, may require more advanced codes/models than commonly used codes such as DandD or RESRAD-ONSITE[13]. Complex sites may also include sites with engineered barrier(s), or with complex hydrogeological conditions such as highly fractured geologic formations. Because of site complexity and variability, there are no standard dose analysis review criteria for these sites.

## I.5.3 Recommended Approach

### *I.5.3.1 Generic Criteria for Selection of Codes/Models*

The generic criteria under this subsection pertain to the NRC staff review of codes/models other than commonly used codes; specifically, those developed or modified by the NRC staff (i.e., other than DandD, RESRAD-ONSITE, and RESRAD–BUILD). The reviewer should use the generic criteria when the codes/models have no readily available documentation of testing, verification, and QA/QC review. In this context, the reviewer should use the following generic criteria in reviewing the codes/models selected for the dose assessment:

- The NRC staff should review the adequacy and completeness of the database available on QA/QC aspects of the code/model. The QA/QC database should be comparable to the NRC's QA/QC requirements (NUREG/BR-0167, "Software Quality Assurance Program and Guideline," issued February 1993 (NRC, 1993b) and NUREG-0856, "Final Technical Position on Documentation of Computer Codes for High Level Waste Management," issued June 1983 (NRC, 1983)). The QA/QC should include information on mathematical formulation, code/model assumptions, consistency of the pathways

[12] Newer versions of the code are also acceptable for use.

[13] RESRAD-OFFSITE 4.0 was released in 2020 and contains more complex source term, hydrogeological, air, and surface water transport models compared to RESRAD-ONSITE.

with the assumed conceptual model(s) used in the code, and accuracy of the software to reflect the model's mathematical formulation and correct representation of the process or system for which it is intended.

- The NRC staff should ensure that the software used for the code is in conformance with the recommendations of the Institute of Electrical and Electronics Engineers (IEEE) Standard 830-1984, "IEEE Guide for Software Requirement Specifications."

- The NRC staff should review the adequacy and appropriateness of the code/model documentation with regard to (1) software requirements and intended use, (2) software design and development, (3) software design verification, (4) software installation and testing, (5) configuration control, (6) software problems and resolution, and (7) software validation.

- For uncommon codes/models, the NRC staff should review code data, including (1) a software summary form, (2) a software problem/change form, (3) a software release notice form, and (4) a code/model user's manual, which covers code technical description, software source code, functional requirements, and external interface requirements (e.g., user interface, hardware interface, software interface, and communication interface), if necessary.

- The NRC staff should review the conceptual model of the selected code to ensure compatibility with the specific site conceptual model, including the pathways and the exposure scenario. The source term assumptions of the selected code should also be compatible with the site-specific source term. The staff may accommodate minor modifications in the source term conceptual model, as long as the basic model assumptions are not violated.

- The NRC staff should verify that the exposure scenario of the selected code is compatible with the intended scenario for the site. For example, models/codes designed for the onsite exposure scenario may not be appropriate for assessment of an offsite exposure scenario.

- The NRC staff should review the selected model/code formulation to account for radionuclide decay and progenies. The code should have proper and timely formulation, as well as linkages of decay products with the receptor location and the transport pathways, via corresponding environmental media.

- The NRC staff should examine the documentation of the selected code/model performance; specifically, test and evaluation, as well as a code comparison with commonly used (accepted) codes and models (e.g., DandD and the RESRAD family of codes). The staff should also review documentation on code/model verification, if available, to support decisions for code acceptance.

- The NRC staff should review code/model features of the sensitivity/uncertainty analysis to account for variability in the selection of input parameters and uncertainty in the conceptual model and multiple options for the interpretation of the system.

### *I.5.3.2 Acceptance Criteria for Selection of Site-Specific Codes/Models*

This issue involves the NRC staff's review of additional requirements supporting the justification for using the conceptual model, the numerical mathematical models, the source term model and its abstraction, and the transport and exposure pathway models.

**CONCEPTUAL MODELS**

The NRC staff review should compare the conceptual model for the site with the conceptual model(s) in the selected code, to ensure compatibility with site-specific physical conditions and pathway assumptions for the critical group receptor.

**NUMERICAL MATHEMATICAL MODELS**

The staff should review the equations used in the code to implement the conceptual model and the numerical links between mathematical models to ensure correctness and consistency. For codes developed or modified by the NRC staff (e.g., DandD, RESRAD-ONSITE, and RESRAD–BUILD), minimal review is needed, because the NRC revised these codes and examined them early for consistency with its QA/QC requirements. For less commonly used codes, or codes developed locally by user(s), the NRC staff should verify the numerical mathematical models, including the numerical links between these models. In this context, the reviewer may examine, if necessary, each mathematical model used for the specific transport-exposure pathway, to ensure that the code is designed for its intended use.

**SOURCE TERM MODELS**

The NRC staff should review the source term model(s) used for the specific site. In this context, the review should include the following source term aspects:

- **Building Occupancy Exposure Scenario Source Term:** The NRC staff should review the HSA and other relevant data on the extent of the source and its depth (e.g., within 1 to 10 millimeters (mm) (0.04 to 0.39 in.) deep into the building surface or more). Based on this review, the reviewer should identify the source as surficial or volumetric and examine the assumptions made for the loose/fixed fractions of the source. The review should address the sources of residual radioactivity on surfaces that are not integral parts of the building (e.g., equipment, pipes, and sewer lines) separately, because the applicable model and exposure scenario could be different. Therefore, source term model assumptions for such surfaces should be reviewed on a case-by-case basis.

  The NRC staff should also review the radionuclide mixture comprising the source and whether a constant ratio is assumed in the dose analysis, as well as determine if surrogate radionuclides are being used. The latter two situations may require additional NRC staff verification of the source definition and review of consistency with the intended final survey methodology.

  The review should also include the use of multiple sources (e.g., multiple rooms). Certain codes may provide an option to define multiple sources in various configurations, such as two to three rooms, with multiple-story buildings. The source term under these conditions allows for source depletion due to open air circulation and common ventilation. For example, the RESRAD–BUILD code model uses two- or three-room models with two- or three-story buildings, allowing for air exchange within the rooms,

and source depletion. The review should include the indoor air-quality model (e.g., building ventilation and infiltration), and the indoor air-concentration model, as well as the adaptation of the air-quality model in the RESRAD–BUILD code, to ensure consistency with the site-specific conditions. The review should verify the input parameters associated with these models. The NRC staff may accept such site-specific source term models after an assessment of the compatibility of the source term model with the conceptual model of the site. The review should also include the physical parameters defining the source term, to ensure consistency with site-specific conditions, and the occupancy parameters, to ensure consistency with the exposure scenario.

- **Resident Farmer Exposure Scenario Source Term:** The NRC staff should examine the licensee's source information to determine if the nature and extent of residual radioactivity is consistent with the model/assumptions in the selected code. The review should include evaluation of the vertical and horizontal extent of residual radioactivity, to verify model assumptions on the area and thickness of the contaminated zone, and to determine if there is subsurface and/or groundwater contamination at the site. The DandD model assumes residual radioactivity is located in surface soils only (approximately 15 cm or less). For contaminated zone thicknesses significantly greater than 15 cm, DandD may underestimate the dose and justification would be needed to use DandD to assess the dose from subsurface residual radioactivity. The contaminated area and its shape should also be assessed to check for possible correction of the area and/or for geometry of the source. Additionally, the NRC staff should determine if credit is taken for a clean cover or a barrier over the contaminated zone and the basis for the modeling assumption. Support for the assumed level of performance of the engineered cover or barrier should be evaluated within the context of institutional control assumptions (e.g., active maintenance for the case where institutional controls are in effect; and only passive performance in the case when institutional controls are assumed to no longer be in effect). Although sensitivity analysis on the timing of failure of institutional controls can be performed, institutional controls should be assumed to fail at time=0 years. The evaluation should consider assumptions regarding the performance and degradation of the engineered covers and barriers during the compliance period (e.g., 1,000 years).

  The NRC staff should also review the physical and chemical form of the source to evaluate the adequacy of the underlying soil leaching model(s) available in the selected code. This review should help assess the source mass-balance model and the transport model within the concerned environmental media. In addition, a review of these source term aspects would help establish consistencies for the selection of relevant parameters. The review should include the horizontal distribution and homogeneity of the source and the variation of source concentration with depth. The NRC staff should use either an upper-bounding value for modeling the thickness or an area-weighted approach to calculate the representative thickness, if this approach does not significantly underestimate the potential dose (e.g., it may not be appropriate to represent an area with a thin vadose zone with a thicker vadose zone based on source thickness averaging, if the thicker vadose zone assumption could lead to longer transport times and lower exposure concentrations due to radiological decay). In certain cases, multiple sources or more advanced subsurface source term modeling may be needed to adequately assess dose; these would be evaluated on a case-by-case basis.

## TRANSPORT MODELS

The transport models simulate transport mechanisms of contaminants from the source to the receptor. The NRC staff should review transport models for consistency and compatibility with respect to (1) the source term, (2) the exposure scenario defined for the critical group receptor, and (3) the simplified conceptual model, which describes site-specific physical conditions. The transport models may include the diffusive and advective transport of contaminants via air, surface water, and groundwater. The transport models can be overly simplified, using simple conservative assumptions such that minimal characterization data would be required to execute the model(s). Transport models can also be very complex, requiring advanced mathematical derivation and extensive site-specific, or surrogate data about the site.

For the building occupancy exposure scenario, the associated transport models (e.g., transport models for ingestion, inhalation, and direct exposure pathways) of the DandD code are simple and conservative. For example, the ingestion pathway depends on the effective transfer rate of the removable surface residual radioactivity from surfaces to hands and from hands to mouth. The inhalation transport model depends largely on mechanical disturbance of the contaminated surface, resuspension of residual radioactivity in the air, and subsequent breathing of contaminated air. The external dose formulation assumes exposure from a nonuniform source of residual radioactivity on the walls, ceiling, and floor of a room. This model was found to be comparable to the infinite plane source for the building occupancy exposure scenario (NRC, 1992).

For the resident farmer exposure scenario, the associated DandD transport models include models of contaminants transport to groundwater, to surface water (e.g., three-box model that relies on transfer of contaminate through leaching), and to air (e.g., through dust mass loading and indoor resuspension). Transport models of contaminants via the air include dust loading, resuspension of contaminated soil, and use of mass loading factor for deposition. Transfer of contaminants from the soil/water to plants, fish, animals, and animal products are calculated using a water use model, along with transfer factors, translocation factors, and bio-accumulation factors. Separate models were used for C-14, and H-3, as described in NUREG/CR-5512, Volumes 1, 2, and 3. The RESRAD-ONSITE model can consider residual radioactivity in surface soils (approximately 15 cm) or thicker contaminated zones, with an idealized cylindrical shape of the contaminated zone[14], and allows for a cover at the top of the contaminated zone, if appropriate.

In general, the NRC's review of the selected code should include transport models and the appropriateness of such models with respect to the site-specific conditions (e.g., area, source, unsaturated zone, and aquifer conditions). In addition, the staff should review, for compatibility and consistency, the transport model assumptions and the generic formulation pertaining to the applicable pathways of the critical group exposure scenario. The extent of the transport model review depends on the familiarity of the NRC staff with these models. Because they developed or modified certain commonly used codes/models (e.g., DandD, RESRAD-ONSITE, and RESRAD-BUILD), the NRC staff is more familiar with them, and they would require less of a review than for less common codes/models developed by users or other parties. The NRC review should also include updated new models or code versions and studies on code/models testing, comparison, and verifications.

[14] RESRAD-ONSITE considers a circular source by default; however, RESRAD-ONSITE has the capability of considering more complicated source geometries.

The RESRAD–BUILD code is more advanced than the DandD code because it employs multiple sources and more advanced particulate air transport models. In other words, each contaminated location may be considered a distinct source. Depending on its geometric appearance, the source can be defined either as a volume, an area, or a point source. The RESRAD–BUILD code depends on erosion of the source and transport of part of its mass into the indoor air environment, resulting in airborne residual radioactivity. The RESRAD–BUILD model differs from DandD because it assumes air exchange among all compartments of the building. In other words, the model assumes that the airborne particulates are being loaded into the indoor air of the compartment and then transported to the indoor air of all compartments of the building. In addition to air exchange between compartments, the indoor air model simulates air exchange between compartments and the outdoor air. Descriptions of models pertaining to indoor air quality, air particulate deposition, inhalation of airborne dust, and ingestion of removable materials and deposited dust, were documented in an Argonne National Laboratory report “ANL/EAD/LD–3” (ANL 1994). The exposure pathways in the RESRAD–BUILD code include (1) the external exposure to radiation emitted directly from the source and from radioactive particulates deposited on the floors and exposure caused by submersion from radioactive particulates, (2) inhalation of airborne radioactive particulates, and (3) ingestion of contaminated material directly from the source, as well as airborne particulates deposited onto the surface of the building.

**EXPOSURE PATHWAY MODELS**

The exposure pathway models pertain to the formulation of the links between the radiological source, the transport of contaminants within environmental media, the critical group receptor location, and behaviors of the receptor that lead to its exposure to residual radioactivity through direct exposure, inhalation, and ingestion of contaminated water, soil, plants, crops, fish, meat, milk, and other dairy products. The NRC staff should review the conceptual model(s) that describe the human behaviors that lead, or control, the amount of receptor exposure. Therefore, the occupational, behavioral, and metabolic parameters describing these models should be reviewed and compared with the default model exposure scenarios and associated parameters. The NRC staff should review exposure model(s) and associated parameters to ensure conservatism, consistency, and comparability with site-specific conditions and exposure scenario assumptions. NUREG/CR–5512, Volumes 1, 2, and 3, provide detailed information on default parameters and approaches for changing parameters in dose modeling analysis.

### *I.5.3.3 Option for Selection of Deterministic or Probabilistic Site-Specific Codes*

Licensees may select either a deterministic analysis or a probabilistic approach for demonstrating compliance with the dose criteria in 10 CFR Part 20, Subpart E. A deterministic analysis uses single parameter values for each variable in the code. In contrast, parameter distributions are specified for uncertain variables in a probabilistic analysis, sets of parameter values are selected via sampling from the specified parameter distributions, many sets of parameter values called realizations are run through the model, and a distribution of results is generated and evaluated. Although deterministic sensitivity analysis can be conducted, a deterministic analysis gives more limited information on the uncertainty in the results, based on uncertainty in the parameter values, and is generally less efficient in identifying important model sensitivities. Therefore, the deterministic approach may require a more elaborate justification of code input parameter values and may require further analysis of doses using upper or lower bounding conditions.

NRC-approved data sets for both DandD and the RESRAD family of codes are for the probabilistic calculation and not the deterministic mode.

Section I.7.3.2 of this appendix contains a detailed description of an NRC staff review for both deterministic and probabilistic analyses.

### *I.5.3.4 Modeling of Subsurface Residual Radioactivity*

For subsurface residual radioactivity (residual radioactivity deeper than approximately 15 cm (6 in.)), the NRC staff should review existing historical site data (including previous processes or practices) and site characterization data, to establish an adequate conceptual model of the subsurface source; specifically, the horizontal and vertical extent of residual radioactivity. Section I.2.3.1 describes approaches for subsurface source term abstraction for dose modeling analysis. In some cases, the licensee may wish to develop multiple DCGLs for surface and subsurface DCGLs. Additional information on integration of dose modeling with radiological surveys for license termination is found in Appendix G.

### *I.5.3.5 Generic Description and Development of DandD*

The DandD code has two default land use exposure scenarios: a building occupancy and a resident farmer exposure scenario. The building occupancy exposure scenario is intended to account for exposure to both fixed and removable residual radioactivity within a building. Exposure pathways included in the building occupancy exposure scenario include external exposure to penetrating radiation, inhalation of resuspended surface residual radioactivity, and inadvertent ingestion of surface residual radioactivity. The resident farmer exposure scenario is intended to account for exposure to residual radioactivity in soil. Resident farmer exposure scenario pathways include the following: external exposure to penetrating radiation; inhalation exposure to resuspended soil; ingestion of soil; and ingestion of contaminated drinking water, plant products, animal products, and fish. The predefined conceptual models within DandD are geared to assessing releases of radioactivity, transport to, and exposure along, these pathways.

For the building occupancy exposure scenario, DandD models external exposure to penetrating radiation as an infinite area source, using surface source dose rate factors from “Federal Guidance Report No. 12: External Exposure to Radionuclides in Air, Water and Soil,” issued September 1993 (EPA 1993). Exposure to inhalation of resuspended surface residual radioactivity is modeled as a linear static relationship between surface residual radioactivity and airborne concentrations. The model accounts for ingrowth and decay. Exposure to incidental ingestion of surface residual radioactivity is modeled with a constant transfer rate.

The generic conceptual models for the resident farmer exposure scenario are more complicated because of the large number of exposure pathways and considerations of release of radioactivity from the source area and transport of radionuclides in the environment. DandD models external exposure from volume soil sources when the person is outside as an infinite slab of residual radioactivity 15 cm (6 in.) thick, using dose rate factors from Federal Guidance Report No. 12 for volume residual radioactivity. When the person is indoors, exposure from external radiation is modeled in a similar manner, except the exposure is assumed to be attenuated through the use of a shielding factor (note: the higher the shielding factor, the lower the assumed attenuation). Exposure through ingestion of contaminated animal and plant products is modeled simply through the use of transfer factors. Instantaneous equilibrium is assumed to occur between radionuclide concentration in the soil and the concentration in plants, and between animal feed and animal products.

The generic source term conceptual model in DandD assumes a constant release rate of radionuclides into the water and air pathways. Release of radionuclides by water is assumed to be downward and a function of a constant infiltration rate, constant contaminant zone thickness, constant moisture content, and equilibrium adsorption. DandD assumes that there are no radioactive gas or vapor releases. Release of radioactive particulates is assumed to be upward, instantaneous, uniform, and a function of a constant particulate concentration in the air and the radioactivity within the soil. Radionuclides in the contaminant zone are assumed to be uniformly distributed in a single soil layer, 15 cm (6 in.) thick. No transport is assumed to occur within the source zone, but radioactive decay is taken into account. In terms of containment, DandD assumes that there are no containers (or that they have failed), and that there is no cover over the contaminated zone.

The DandD generic conceptual model for the groundwater pathway assumes a single hydrostratigraphic layer for each of the unsaturated and saturated zones. The unsaturated zone (vadose zone) can be broken into multiple layers within DandD; however, each layer is assumed to have the same properties. For radionuclides entering the vadose zone, DandD accounts for adsorption-limited leaching by considering the vadose zone to behave as a well-mixed chemical reactor with a constant water inlet and outlet rate set at the infiltration rate. Accordingly, it is assumed that the vertical saturated hydraulic conductivity of the unsaturated zone is greater than or equal to the infiltration rate (i.e., there is neither ponding nor runoff on the surface). The outlet concentration from one unsaturated zone layer to another is assumed to be a function of the constant infiltration rate, equilibrium partitioning, the thickness of the layer, a constant moisture content, and radioactive decay. Radionuclides entering the saturated zone are assumed to be instantaneously and uniformly distributed over a constant volume of water equivalent to the larger of either the volume of infiltrating water (i.e., the infiltration rate times the contaminated area) or the sum of the water assumed to be removed for domestic use and irrigation. Based on the default parameters in DandD, dilution in the groundwater pathway is based on the water use. No retardation is assumed to occur in the aquifer; however, radioactive decay is taken into account. A volume of contaminated water equivalent to the irrigation volume is assumed to be returned annually to the source zone. The concentration of radionuclides in the irrigation water is assumed to remain constant during the year. Radionuclides deposited on the vegetation are assumed to be removed at a constant rate. The DandD groundwater model should generally provide a conservative representation of the groundwater system, because it allows very little dilution and nominal attenuation.

The generic surface water conceptual model in DandD assumes that radionuclides are uniformly mixed within a finite volume of water representing a pond. Radionuclides are assumed to enter the pond at the same time and concentration as they enter the groundwater. Accordingly, there is assumed to be no transport of radionuclides through the groundwater to the pond and thus no additional attenuation (besides the initial groundwater dilution) is assumed for transport in the groundwater. The surface water model within DandD should provide a conservative dose estimate as long as a small volume is assumed for the surface water pond. Because the parameters in DandD are selected to provide a conservative dose estimate, the generic conceptualization of the surface water pathway should generally provide a conservative representation of transport of radionuclides through it. Figure I.7 shows the generic groundwater and surface water conceptual model within DandD.

The generic conceptual model of the air pathway in DandD assumes an equilibrium distribution between radionuclides in the air and soil. The concentration in air is assumed to be a function of the soil concentration and a constant dust loading in the air. Accordingly, all radionuclides in the air are assumed to be in a particulate form. The air pathway model within DandD is very

simple and should generally allow a conservative dose estimate, as long as a conservative particulate concentration is assumed. Because the default parameters in DandD are geared to be conservative, the air pathway in DandD should generally allow a conservative dose estimate.

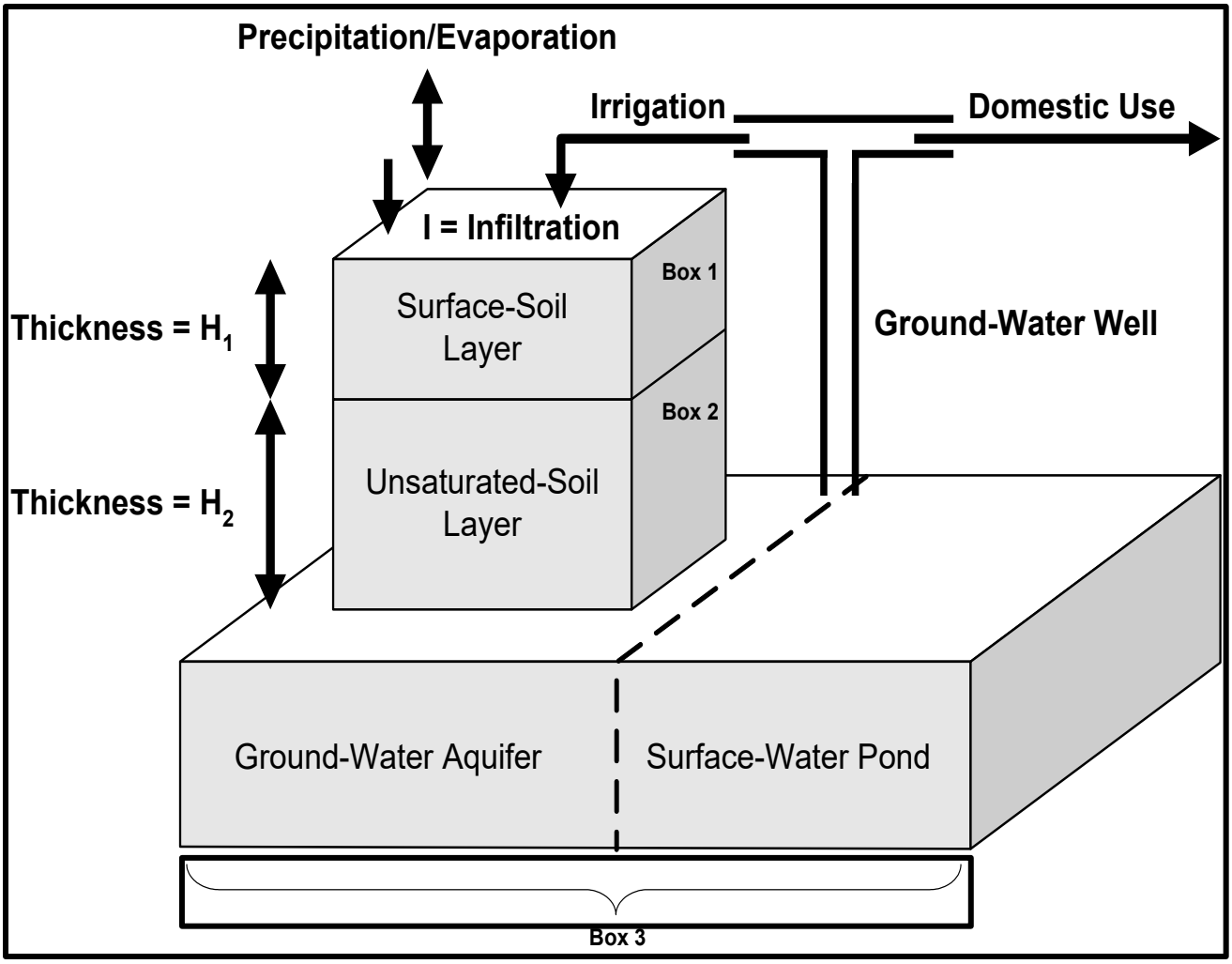


**Figure I.7 DandD Conceptual Model of the Groundwater and Surface Water Systems (from NUREG/CR-5621)**

## PROBABILISTIC DANDD VERSION 2

The NRC staff developed a probabilistic DandD Version 2, which updates, improves, replaces and significantly enhances the capabilities of Version 1.0. In particular, Version 2 allows full probabilistic treatment of dose assessments, whereas Version 1.0 embodied constant default parameter values and only allowed deterministic analyses. DandD implements the methodology and information contained in NUREG/CR–5512, Volume 1 (NRC, 1992), as well as the parameter analysis in Volume 3 (NRC, 1999c), which establishes the PDFs for all of the parameters associated with the exposure scenarios, exposure pathways, and models embodied in DandD.

Finally, DandD Version 2 includes a sensitivity analysis module that assists licensees and the NRC staff to identify those parameters in the screening analysis that have the greatest impact on the results of the dose assessment. Identification of risk-significant parameters helps licensees make informed decisions on the allocation of resources needed to gather site-specific information important to the compliance demonstration.

### *I.5.3.6 Generic Description of the RESRAD Family of Codes*

Argonne National Laboratory developed the RESRAD family of computer codes under the sponsorship of the U.S. Department of Energy, and other agencies, such as the NRC. These codes are pathway analysis models designed to evaluate potential radiological doses associated with exposure of members of the public to residual radioactivity in soils and building materials, respectively. The RESRAD-ONSITE code uses a residential farmer exposure scenario (ANL/EAD/LD–2, "Manual for Implementing Residual Radioactive Material Guidelines Using RESRAD 5.0," issued September 1993 (ANL, 1993a)) with nearly identical exposure pathways as the DandD residential exposure scenario described in NUREG/CR–5512, Volume 1 (NRC, 1992). The RESRAD–BUILD code uses a scenario that covers all exposure pathways in the DandD building occupancy exposure scenario, plus pathways corresponding to external exposures from air submersion and deposited material and to ingestion of deposited material. Previous sections in this appendix briefly described the RESRAD and RESRAD–BUILD codes and conceptual models (see Section I.4.3.2). The RESRAD-OFFSITE code extends RESRAD-ONSITE capabilities to include offsite exposure locations, as well as including more complex source terms, groundwater, air and surface water models. Detailed descriptions of these codes are available in various technical documents, including:

- ANL/EAD/LD-2, "Manual for Implementing Residual Radioactive Material Guidelines Using RESRAD, Version 5.0," issued September 1993 (ANL, 1993a);
- ANL/EAD/LD-3, "RESRAD-BUILD: A Computer Model for Analyzing the Radiological Doses Resulting from the Remediation and Occupancy of Buildings Contaminated with Radioactive Material," issued November 1994 (ANL, 1994);
- NUREG/CR-6697, "Development of Probabilistic RESRAD 6.0 and RESRAD-BUILD 3.0 Computer Codes," issued December 2000 (NRC, 2000c);
- NUREG/CR-7268, "User's Manual for RESRAD-OFFSITE Code Version 4, Vol. 1—Methodology and Models Used in RESRAD-OFFSITE Code," also ANL/EVS/TM-19/2, Volume1, issued February 2020 (NRC, 2020b)
- NUREG/CR-7268, "User's Manual for RESRAD-OFFSITE Code Version 4, Vol. 2—User's Guide for RESRAD-OFFSITE Version 4," also ANL/EVS/TM-19/2, Volume 2, issued February 2020 (NRC, 2020c).

The NRC staff and licensees widely used the deterministic versions of these codes before the LTR to estimate doses from radioactively contaminated sites and structures. The NRC sponsored development of the probabilistic versions (RESRAD-ONSITE, Version 6, and

RESRAD-BUILD, Version 3), and their default probabilistic data sets. These two codes[15] were selected because they possess all three of the following attributes:

(1) The software has been widely accepted, and there is already a large user base among the NRC staff and licensees.

(2) The models in the software were designed, and have been applied successfully, to more complex physical and residual radioactivity conditions than the DandD code.

(3) Verification and validation of these two codes are well documented (C. Yu, "RESRAD Family of Codes and Comparison with Other Codes for Decontamination and Restoration of Nuclear Facilities," issued 1999 (Yu, 1999)).

It should be noted that the RESRAD-ONSITE code has been widely used and tested by national and international agencies and has gone through verification (HNUS-ARPD-94-174, "Verification of RESRAD—Case for Implementing Residual Radioactive Material Guidelines, Version 5.03," issued June 1994 (HNUS, 1994)), dose model comparison (NRC, 1999d; Electric Power Research Institute (EPRI) TR–112874, "Comparison of Decommissioning Dose Modeling Codes for Nuclear Power Plant Use: RESRAD and DandD," issued November 1999 (EPRI, 1999), and benchmarking (DOE/ORO–2033, "Benchmarking Analysis of Three Multimedia Models: RESRAD, MMSOILS and MEPAS," issued November 1995 (DOE, 1995)). RESRAD-ONSITE, RESRAD-OFFSITE, and RESRAD–BUILD codes are continuously being developed and updated with new code versions. Licensees may use updated versions of the RESRAD family of codes that have gone through adequate verification, validation, and QA testing and should document the version of the codes used in their DP.

**RESRAD–BUILD**

The RESRAD–BUILD code can be used to evaluate doses for the building occupancy exposure scenario. It considers exposure from external radiation at the source and air submersion, inhalation of airborne material, and inadvertent ingestion of radioactive material. Exposure to direct radiation at the source is calculated using surface source dose rate factors from Federal Guidance Report No. 12. The RESRAD–BUILD code incorporates correction factors to account for a finite area source, for any offset of the receptor from the axis of the disk of residual radioactivity, and for shielding by building materials. Exposure to external radiation from air submersion is calculated as an infinite cloud of material using dose rate conversion factors for an infinite cloud. The RESRAD–BUILD code models the airborne concentration of radionuclides using a dynamic model that accounts for the kinetic introduction and removal of radioactive material to and from indoor air. Exposure to incidental ingestion of radioactive material is modeled using a constant transfer rate.

**RESRAD-ONSITE AND RESRAD-OFFSITE**

The RESRAD-ONSITE code can be used for analyzing the resident farmer exposure scenario. As with the generic conceptual models used by DandD for analyzing the resident farmer and building occupancy exposure scenarios, the conceptual model in RESRAD-ONSITE (see Figure I.8) is more complex than the exposure scenario in RESRAD–BUILD. The RESRAD-ONSITE code considers external exposure from volume soil sources when the receptor is

[15] RESRAD-OFFSITE was developed after the development of the probabilistic versions of RESRAD-ONSITE and RESRAD-BUILD.

outside, using volume dose rate factors from Federal Guidance Report No. 12. Correction factors are used to account for soil density, areal extent of residual radioactivity, thickness of residual radioactivity, and cover attenuation. When the person is indoors, exposure from external radiation is modeled in a similar manner, except that additional attenuation is considered to account for protection afforded by the building. Exposure through ingestion of contaminated animal and plant products is modeled simply through use of transfer factors. RESRAD-OFFSITE extends RESRAD-ONSITE to offsite locations and features more complex source term; and atmospheric, groundwater, and surface water transport models. Additional details regarding each of the models is found below.

*Source Term Models*

The generic source term conceptual model in RESRAD-ONSITE assumes a time-varying release rate of radionuclides into the water and air pathways.[16] Radionuclides in the contaminated zone are assumed to be uniformly distributed. No transport is assumed to occur within the source zone, but radioactive decay is accounted for. In terms of containment, the radioactive material is not assumed to be contained (or containers are assumed to have failed).[17] RESRAD-ONSITE allows inclusion of a cover over the contaminated area. However, the cover is not assumed to limit infiltration of water. The cover provides shielding from gamma radiation and can also decrease the amount of radioactivity taken up by plants. Release of radionuclides by water is assumed to be a function of a constant infiltration rate, time-varying contaminant zone thickness, constant moisture content, and equilibrium adsorption. The contaminant zone is assumed to decrease over time from a constant erosion rate. RESRAD-ONSITE assumes a uniform release of H-3 and C-14 gases, based on a constant evasion loss rate. Particulates are assumed to be instantaneously and uniformly released into the air as a function of the concentration of particulates in the air, based on a constant mass loading rate.

Figure I.9 provides conceptual information on the impact of the $K_d$ on the magnitude of source release from the source zone and the mobility of contaminants in the natural environment, as conceptualized in RESRAD-ONSITE. Zones 1 and 2 depicted in Figure I.9 are the contaminated and unsaturated zones, respectively. For relatively low contaminated zone $K_d$s, RESRAD-ONSITE predicts a higher concentration release into the environment, signified by the dark blue contaminated water leaching into the unsaturated zone (Cases A and B). For relatively high contaminated zone $K_d$s, RESRAD-ONSITE predicts a lower concentration release into the environment, signified by the light blue contaminated water leaching into the unsaturated zone (Cases C and D). The $K_d$ assigned to the unsaturated zone controls transport times to the aquifer with lower unsaturated zone $K_d$s leading to faster transport rates or plumes located lower in the unsaturated zone (Cases A and C) and higher unsaturated zone $K_d$s leading to slower transport rates or plumes located higher in the unsaturated zone (Cases B and D) after the same amount of time has elapsed. The $K_d$ is used to calculate the leach rate using Equations 3 and 4. The RESRAD-ONSITE code uses the unsaturated zone distribution coefficient to calculate travel times from the contaminated zone to the saturated zone.

[16] Additional source term options are available in RESRAD-OFFSITE 4.0 (NRC, 2020b).

[17] RESRAD-OFFSITE allows specification of the fraction of residual radioactivity that is releasable over time.

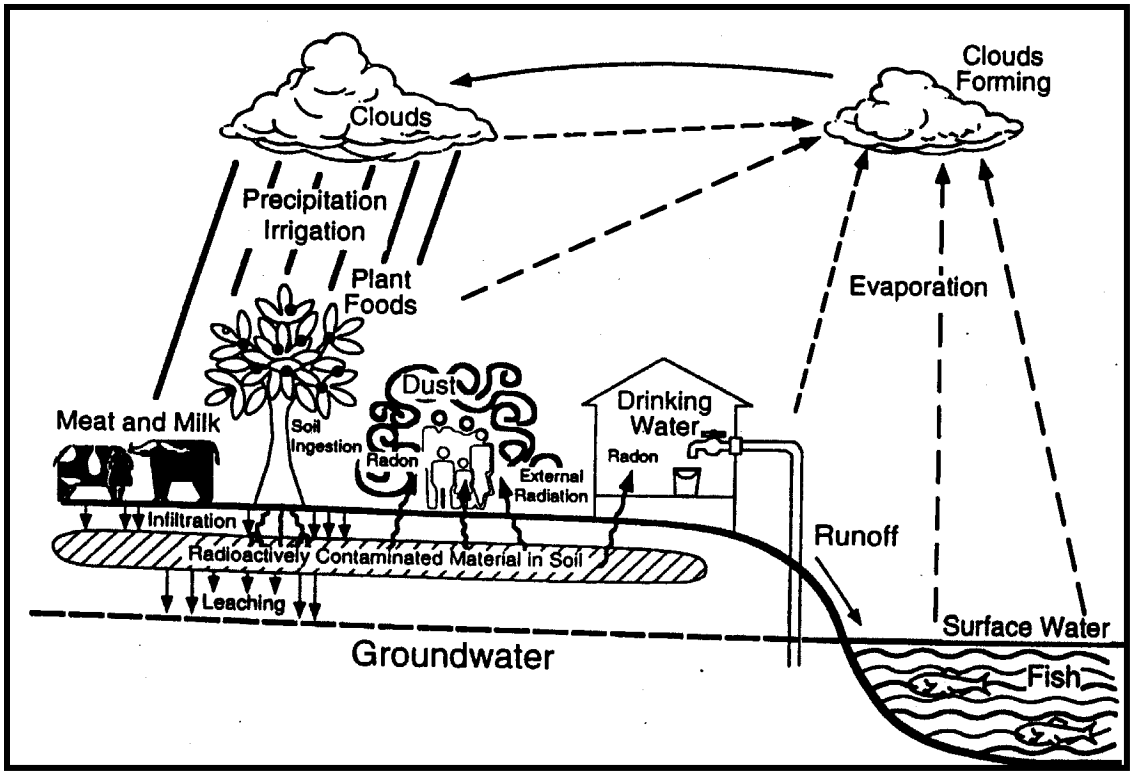


**Figure I.8 Conceptualization Modeled by RESRAD-ONSITE (ANL, 1993)**

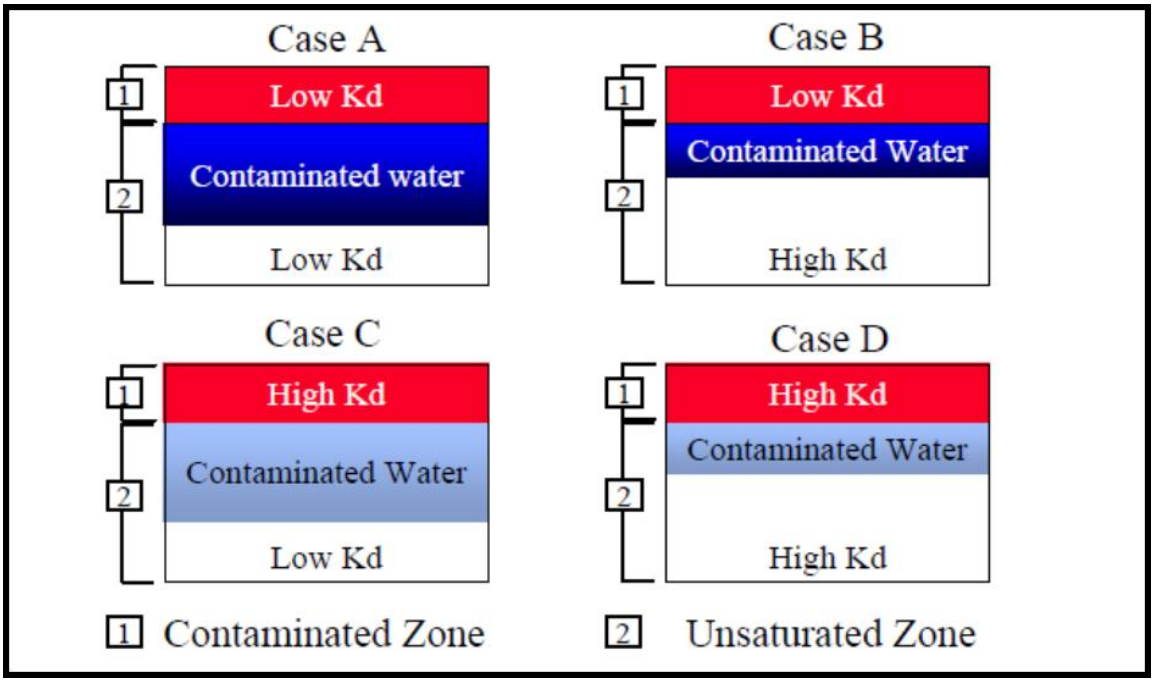


**Figure I.9 RESRAD Near-Field Leaching and Transport Model (from RESRAD Training Workshop)**

The RESRAD-ONSITE code calculates a leach rate based on a user-specified $K_d$ for the contaminated zone.[18] Alternatively, the user may specify (1) time since material placement and groundwater concentrations, (2) solubility limits, or (3) a leach rate from which contaminated zone $K_d$s can be back-calculated for individual radionuclides, or allows for the correlation of plant to soil coefficient to $K_d$.[19] Only one of the methods can be used and, if more than one user-specified approach is indicated, the code selects the approach first specified in the order listed above. Ultimately, all options use the same approach to characterizing the source term from the contaminated zone, based on $K_d$ and a calculated leach rate.

$$L_i = \frac{I}{\theta^{(cz)} \, x \, T_o \, x \, R_{d_i}^{(cz)}} \qquad \text{Equation 3}$$

Where

$I$ = infiltration rate (m/y),

$\theta^{(cz)}$ = volumetric water content of the contaminated zone,

$T_o$ = initial thickness of the contaminated zone (m), and

$R_{d_i}^{(cz)}$ = retardation factor in the contaminated zone for radionuclide i (dimensionless).

The retardation factor is calculated as follows:

$$R_{d_i} = 1 + \frac{\rho_b}{\theta} K_{d_i} \qquad \text{Equation 4}$$

Where

$\rho_b$ = bulk soil density (g/cm$^3$),

$K_{d_i}$ = distribution coefficient for the ith principal radionuclide (cm$^3$/g), and

$\theta$ = volumetric water content (dimensionless).

The $K_d$ is often one of the single most important parameter values in dose modeling, as it dictates the release rate of radionuclides out of the source or contaminated zone. Therefore, discussion on the importance of this parameter used in source term modeling is warranted. To estimate release rates and concentrations of radionuclides in groundwater, an analyst needs some means of quantifying the relationship between dissolved contaminants and the solid phase through which the contaminants move. In many cases, dose analysts resort to the use of $K_d$s,[20] because the $K_d$ model is readily available in commonly used contaminant transport models, and data are typically lacking to perform more complex modeling.

Simply put, $K_d$s characterize the partitioning of contaminants between the solid and aqueous phases. The solid phase concentration is linearly related to the aqueous phase concentration by the $K_d$. Higher $K_d$s indicate a greater degree of partitioning or "sorption" of a contaminant to solid materials. The term "sorption" is used when the actual mechanism of chemical removal from the aqueous phase is unknown. For example, sorption can refer to ion exchange, in which ions are loosely associated with a surface through differences in electrical charge, typically on porous materials with fixed charges, such as clays, zeolites, and organic material. Sorption also refers to surface complexation (more properly termed adsorption) in which the ion in solution reacts with ions on a solid's surface in a way that is analogous to complex forming reactions that

[18] RESRAD-ONSITE allow for separate specification of $K_d$s for the unsaturated and saturated zones.

[19] The user has the option of toggling this option on or off.

[20] Distribution coefficient and $K_d$ are used interchangeably in this desk guide.

occur in solution. In both of these cases, sorption is subject to influences by the solution chemistry, notably pH and ionic strength. Changes in solution chemistry can, therefore, lead to a reversal in sorption and desorption. This is a significant consideration in long-term assessments, where water chemistry cannot necessarily be expected to remain constant, potentially resulting in sorption of a contaminant for a while, and then its release as water chemistry changes. Therefore, the potential for changes in geochemical conditions is an important consideration in decommissioning dose modeling.

Owing to its simplicity, the $K_d$ model is widely applied in contaminant transport codes. Any computer code used to simulate the transport of a conservative, nonreactive constituent can be easily modified to solve the transport equation for a reactive (sorbing) constituent, if the partitioning coefficient is assumed to be constant (Glynn 2002)). It is expected that, for the vast majority of decommissioning sites, use of a $K_d$ model is adequate. However, the selection of an appropriate value or set of values to use in performance assessment modeling can be a challenging task. Furthermore, the $K_d$ model inherently assumes certain conditions that may not be valid: linearity of the sorption isotherm, equilibrium conditions, and the existence of only a single aqueous and solid phase species (simple and invariant sorption). If saturation of sorption sites, rapid transport, or variable chemical conditions is a concern, an analyst should carefully evaluate the appropriateness of using a $K_d$ model to evaluate site risk.

If the $K_d$ appears to be inadequate to describe the release rate of radionuclides from the contaminated zone, RESRAD-ONSITE also allow editing of a release rate table that provides the user with additional flexibility in altering the source term model used by the code. In addition to the source term model offered in RESRAD-ONSITE, RESRAD-OFFSITE has the ability to delay release and to consider the fraction of the residual radioactivity that is releasable over time (e.g., to simulate the use of containers or engineered barriers). The RESRAD-OFFSITE code also provides options to account for first order rate controlled release with transport through the contaminated zone, as well as an instantaneous equilibrium desorption release option (see NUREG/CR-7127, “New Source Term Model for the RESRAD-OFFSITE Code Version 3,” issued December 2013 (NRC, 2013) and NUREG/CR-7189, “User’s Guide for RESRAD-OFFSITE,” issued April 2015 (NRC, 2015b) for additional details). RESRAD-OFFSITE 4.0 was released in early 2020 and offers additional source term models including solubility-limited and diffusive release models (NRC, 2020a; and NRC, 2020b). If a more complex source term is needed than the first order leach rate models used in DandD and RESRAD-ONSITE, the licensee may consider use of a more complex model offered in RESRAD-OFFSITE, or other codes such as DUST-MS (Sullivan, 1996).

As a result of the NRC staff review of dose modeling to support decommissioning and performance assessment modeling to support low-level waste disposal, a number of observations and recommendations are noted with respect to the use of $K_d$s in the abstraction of the source term and groundwater models. Table I.8 lists key considerations.

**Table I.8 Distribution Coefficient Model and Model Simplification Considerations**

| Category | Consideration or Recommendation |
|---|---|
| **Model** | Extrapolation of experimental data to fit a linear $K_d$ model could lead to significant inaccuracies in concentration and dose projections. For example, if experiments are conducted at relatively low concentrations, the calculated $K_d$ may not be appropriate for higher concentrations. Saturation of sorption sites at higher concentrations may lead to nonlinearity in the sorption isotherm (i.e., less sorption may occur at higher concentrations). In another example, if experiments are conducted at relatively high concentrations, solubility limits may be exceeded and calculated $K_d$s may be overestimated. |
| | A particular $K_d$ value or the $K_d$ model, in general, may be inappropriate if nonequilibrium conditions are expected in the field. For example, under certain conditions (e.g., transport rates are rapid relative to sorption reaction rates, as may be the case under conditions of rapid fracture flow), sorption may be overestimated and dose/risk underestimated. For batch experiments, $K_d$s may increase with equilibration time. However, if nonequilibrium conditions occur in the field, doses and risk could be significantly underestimated. Experiments used to derive site-specific $K_d$s should be representative of field conditions. Column experiments could be run to study the impact of pore water velocity on sorption rates. |
| | Use of a single $K_d$ will likely be inappropriate under variable geochemical conditions. Methods to overcome this problem include use of spatially and temporally variable $K_d$s, $K_d$ lookup tables, or the establishment of functional relationships between $K_d$ and other key parameters. |
| **Model Simplifications** | Averaging $K_d$s of different species of the same radioisotope is generally not appropriate and can lead to significant errors in modeling results. In limited cases, the benefits of $K_d$ averaging may outweigh the costs (costs such as calculation or decision error). However, adequate justification should be provided to support the approach, and reviewers should carefully evaluate the justification. |
| | In some cases, errors associated with the averaging of $K_d$s of different materials for a single species may be acceptably low. However, if differences in material $K_d$s are significant, peak contaminant breakthrough times could be markedly different and lead to large inaccuracies in modeling predictions, particularly for relatively short-lived radionuclides. Adequate justification should be provided for any material $K_d$ averaging approach and should include comparisons of results for the full range of parameter space being simulated. Inaccuracies related to timing of peak dose should be quantified and clearly communicated. |

*Groundwater Models*

The RESRAD-ONSITE generic conceptual groundwater model assumes one or more horizontal homogeneous strata for the unsaturated zone. Transport in the unsaturated zone is assumed to result from steady-state, constant vertical flow, with equilibrium adsorption, and decay, but no dispersion. The RESRAD-ONSITE code has two different ways of modeling radionuclides once they reach the saturated zone. In the mass-balance approach, radionuclides entering the saturated zone are assumed to be instantaneously and uniformly distributed over a constant volume equivalent to the volume of water removed by the hypothetical well (as long as the pumping rate is larger than the rate of leachate entering the groundwater—if not, no dilution is assumed to occur in the groundwater). For the mass-balance approach, radionuclides are assumed to enter a well pumping immediately beneath the contamination zone. The mass-balance approach is very similar to the groundwater modeling approach in DandD. In the nondispersion approach, transport in the saturated zone is assumed to occur in a single homogeneous stratum, under steady-state, unidirectional flow, with a constant velocity, equilibrium adsorption, and decay. It assumes no dispersion; however, radionuclides are assumed to be diluted by clean water as a function of the assumed capture zone of the hypothetical well, in relation to the width of residual radioactivity and the depth of residual radioactivity, in relation to the depth of the hypothetical well (see Table I.9 below). Radioactive decay and equilibrium adsorption are assumed to occur for the nondispersion approach. Radionuclides are assumed to enter a well located at the immediate downgradient edge of the contamination zone. For the nondispersion model, the calculated width of the effective pumping zone could be a factor of 2 larger than what one would predict from a steady-state capture zone analysis; this could lead to a slight overestimation in the amount of dilution (NRC, 1999d).

**Table I.9 Four Cases Evaluated in RESRAD's Nondispersion Model to Calculate Dilution**

| Case | Dilution Factor | Condition | Condition | Parameters |
|---|---|---|---|---|
| 1 | 1 | $d_r \le A / l$ | $\zeta \ge d_w$ | $d_r$=effective pumping width (m)<br>A=contaminated zone area<br>*l*= length of the contaminated zone parallel to the hydraulic gradient<br>ζ=distance from the water table to the lower boundary of the contaminant plume<br>$d_w$=distance of well intake below the water table<br>$U_w$ = well pumping rate |
| 2 | $A * I / U_w$ | $d_r > A / l$ | $\zeta < d_w$ | |
| 3 | $A * I * d_w / U_w * \zeta$ | $d_r > A / l$ | $\zeta \ge d_w$ | |
| 4 | $\zeta / d_w$ | $d_r \le A / l$ | $\zeta < d_w$ | |

In determining which of these two conceptual models to use, the licensee should consider where the hypothetical well may be located (i.e., either at the center of the residual radioactivity or at the edge of the residual radioactivity), the relative half-life of the radioactivity, and the potential capture zone of the hypothetical well. Use of the nondispersion model will generally result in lower estimated doses. Both models assume that radionuclides enter the well as soon as they reach the water table. However, the nondispersion model, unlike the mass-balance model, calculates the time it takes for the peak concentration to occur after the initial breakthrough. Accordingly, the nondispersion model accounts for radioactive decay during the interval between the initial breakthrough and the arrival of the peak concentration. Generally, the amount of decay should be small unless the radionuclides have short half-lives and are retarded. In addition, unlike with the mass-balance model, no assumption is made for the

nondispersion model that all radionuclides released from the contaminated zone are withdrawn through the well. Therefore, the nondispersion model may include additional dilution. The only way that dilution is not considered is if the expected capture zone of the hypothetical well is small in relation to the width and thickness of the residual radioactivity and the plume is deeper than the well intake depth. Because the nondispersion model will generally give a lower estimated dose than the mass-balance model, it is important for the licensee to justify the use of this model for the specific analysis. Use of the mass-balance approach should always be acceptable. In Equations 5 and 6, use of the nondispersion model should be acceptable, without additional justification, for modeling long-lived radionuclides (i.e., where radioactive decay is not important) when either one of the following conditions is met:

$$\frac{U_w}{v \bullet d_w} > \frac{A}{len} \quad \text{and}$$

$$\left(\frac{I}{v}\right) len < d_w$$

Equation 5

$$\frac{U_w}{v \cdot d_w} \leq \frac{A}{len} \quad \text{and}$$

$$\left(\frac{I}{v}\right) len \geq d_w$$

Equation 6

where $U_w$ = pumpage rate from the well ($m^3/y$);
v = groundwater darcy velocity (m/y);
$A$ = area of residual radioactivity ($m^2$);
$d_W$ = depth of well intake below water table (m);
*len* = length of residual radioactivity parallel to groundwater flow (m); and
$I$ = infiltration rate (m/y).

As a general rule, use of the nondispersion approach should be acceptable when the area of residual radioactivity is known to be larger than the assumed capture area of the hypothetical well. Assuming an essentially flat water table and steady-state conditions, the capture area of the hypothetical well can be calculated in Equation 7 as follows:

$$A_w = \left( \frac{U_w}{I} \right) \qquad \text{Equation 7}$$

where $A_W$ = area of well capture ($m^2$);
$U_W$ = pumpage rate from the well ($m^3$/y); and
$I$ = infiltration rate (m/y).

The RESRAD-OFFSITE code provides a more complex model for simulating groundwater transport to offsite locations. The RESRAD-OFFSITE code can consider advection and dispersion in calculating groundwater concentrations at a receptor well. The groundwater transport model in RESRAD-OFFSITE considers 1-D advection (straight or curved flow path), along with 3-D dispersive transport in the saturated zone. Likewise, while only 1-D advection is considered in RESRAD-ONSITE, RESRAD-OFFSITE considers 1-D advection and 1-D dispersive transport in the unsaturated zone. Furthermore, the unsaturated zone, saturated zone, and contaminated zone[21] can be subdivided into smaller zones to increase the accuracy of transport simulations.[22]

Both RESRAD-ONSITE and RESRAD-OFFSITE have the capability to consider variable transport rates of progeny created during transport in groundwater. The RESRAD-OFFSITE code has two groundwater transport algorithms: the first algorithm considers variable transport rates of parents and progeny, and the second algorithm models longitudinal dispersion[23]. When either variable transport rates or dispersion is clearly dominant, the RESRAD-OFFSITE user should choose the transport algorithm that is most important. Specification of the most important transport mechanism is important to increasing computational efficiency. When both the longitudinal dispersion and the variable transport rates are important, the user has the option of subdividing the transport pathway into a number of subzones to more accurately simulate the transport of progeny in transport, although this approach may significantly increase computation times. Only the zone where the progeny atoms are created would not consider both processes (longitudinal dispersion and variable transport rates of parents and daughters).

Code developers benchmarked RESRAD-OFFSITE against RESRAD-ONSITE (C.E. Yu, "Benchmarking of RESRAD-OFFSITE: Transition from RESRAD (onsite) to RESRAD-OFFSITE and Comparison of the RESRAD-OFFSITE Predictions with Peer Codes," issued May 2006) (Yu, 2006). Benchmarking was performed with an earlier version of the RESRAD-OFFSITE code (i.e., benchmarking was conducted before the 2007 release of RESRAD-OFFSITE Version 2.0). The results of the benchmarking exercises showed that RESRAD-OFFSITE could mimic the results of RESRAD-ONSITE when certain parameters were changed consistent with the RESRAD-ONSITE conceptual model (onsite dose). Notable differences between initial simulations run with RESRAD-ONSITE and RESRAD-OFFSITE included travel times to the point of compliance that were attributable to differences in the use of porosity in the transport calculations (i.e., effective porosity is used in RESRAD-ONSITE while total porosity is used in

[21] The capability to more accurately simulate transport through the contaminated zone through use of subzones was added in RESRAD-OFFSITE Version 3.0.

[22] It is important to note that RESRAD-OFFSITE also has the capability of mimicking the RESRAD-ONSITE code for calculation of doses to an onsite receptor. However, reference to RESRAD-OFFSITE models and calculations in this section pertains to just the offsite capabilities and not the onsite dose calculations, unless otherwise stated.

[23] Both RESRAD-OFFSITE transport algorithms account for transverse dispersion.

RESRAD-OFFSITE). Another noteworthy difference in results was observed for the water-dependent pathways due to accumulation of radioactivity in soil from applying contaminated irrigation water that is considered in RESRAD-OFFSITE but is not considered in RESRAD-ONSITE.

*Surface Water Models*

The generic conceptual model of the surface water pathway in RESRAD-ONSITE assumes that radionuclides are uniformly distributed in a finite volume of water within a watershed. Radionuclides in surface water are assumed to be diluted as a function of the size of the contaminated area in relation to the size of the watershed. Radionuclides are assumed to enter the watershed at the same time as the groundwater. Accordingly, no additional attenuation is considered as radionuclides are transported to the watershed. The RESRAD surface water conceptual model assumes that all radionuclides reaching the surface water are derived from the groundwater pathway. Thus, neither the transport of radionuclides overland from runoff nor additional dilution from overland runoff is considered.

A more complex surface water model was added to RESRAD-OFFSITE 4.0 released in early 2020 (NRC, 2020a; and NRC, 2020b). In addition to contributions of residual radioactivity from groundwater to surface water, this model considers contributions of residual radioactivity primary contamination and catchment areas to surface water from surface water runoff and erosion, and deposition of particulates from atmospheric transport. Various sources and sinks of residual radioactivity are also considered in this model.

For example, the water balance considers the interception of groundwater by the surface water body, inflow of runoff and stream flow from the catchment into the surface water body, precipitation on the surface water body, evaporation from the surface water body, stream flow and extraction out of the surface water body, and infiltration from the surface water body into the groundwater aquifer. The sediment balance considers the influx of eroded material, removal of sediments out of the surface water body due to flow, settling of suspended sediment, and other processes. Sources of radioactivity to the surface water include

- discharge from surface water,
- eroded particulates from the primary contamination,
- surface water runoff from the catchment (also includes particulates deposited in the catchment through the air pathway), and
- particulates deposited directly on the surface water body through the air pathway

Sinks include mass removal due to stream flow, infiltration from the surface water body to the aquifer, and burial of radioactivity in deeper sediments, radiological decay, and other processes. Equilibrium partitioning of radionuclides between the water, the suspended sediments and the bottom sediments is also assumed and represents a removal mechanism from the aqueous phase (Yu et al., 2019).

*Air Models*

The generic conceptual model of the air pathway in RESRAD-ONSITE use a constant mass loading factor and area factor to model radionuclide transport. The area factor, which is used to

estimate the amount of dilution, relates the concentration of radionuclides from a finite area source to the concentration of radionuclides from an infinite area source. It is calculated as a function of particle diameter, wind speed, and the side length of a square-area source. The conceptual model assumes a fixed particle density, constant annual rainfall rate, and constant atmospheric stability. No radioactive decay is considered. Chang, et al. (1998) provides more detail. Tritium and C-14 gases are assumed to be uniformly mixed in a constant volume of air above the contaminated zone. RESRAD-ONSITE does not model the transport of H-3 or C-14 as particulates in the air.

A more complex atmospheric dispersion model is utilized in RESRAD-OFFSITE compared to RESRAD-ONSITE. The air dispersion model in RESRAD-OFFSITE uses a polar grid with 16 sectors to specify wind direction. Meteorological inputs include joint frequency of wind speed and stability class for each of the 16 sectors. Pasquill-Gifford, Briggs rural and Briggs urban dispersion coefficients can be selected for dispersion model coefficients. Sector-averaged, ground-level air concentrations are calculated by the code for various distances down-wind of the source. Removal processes include dry and wet deposition, as well as radiological decay.

Table I-10 provides summary information on the attributes and information requirements of the DandD versus RESRAD-ONSITE and RESRAD-BUILD computer codes. Additional information on the differences between the codes can be found in NUREG/CR-5512, Volume 4 (NRC, 1999d).

### I.5.4 Use of Codes and Models Other than DandD and the RESRAD Family of Codes

The NRC staff should provide flexibility for the possible use of other codes and models selected by licensees. However, less common codes, specifically those developed by users, may require more extensive NRC staff review and verification. In this context, the NRC may review the following pertinent aspects when using other less common codes:

- scope of code application and applicability to the concerned site
- extensive review of the generic code selection criteria listed previously
- review of the mathematical formulation of the associated models and the selected dose conversion factors
- review of the conceptual model, including the source term model, used in the code, and compatibility with site conditions
- review of code performance and comparison with commonly used and verified codes
- review of code capability on the handling of default pathways and consistency in selecting default parameters (e.g., occupancy, behavioral, and metabolic parameters)
- detailed review of code/model documentation and updates for code/model modifications, including QA/QC reviews

**Table I.10 Summary Attributes and Information Needs of DandD; and RESRAD-ONSITE and RESRAD-BUILD Codes**

| | DandD | RESRAD-ONSITE and RESRAD-OFFSITE | RESRAD-BUILD |
|---|---|---|---|
| **General** | • Can be used for screening or site-specific analyses for buildings and soils.<br>• Code QA/QC has already been performed. | • Used for site-specific analyses for soils.<br>• Code QA/QC has already been performed. | • Used for site-specific analyses for buildings.<br>• Code QA/QC has already been performed. |
| **Conceptual and Mathematical Models** | • Conceptual site model must be compatible with the assumptions in the code. For soils, the residual radioactivity is assumed to be associated with surface soils, and no radioactivity is assumed to be present in the subsurface or in groundwater. A list of site conditions that may be incompatible with DandD is listed in Table I.5.<br>• For buildings, residual radioactivity is assumed to be associated with building surfaces.<br>• DandD cannot easily handle multiple sources or sources with odd geometry (e.g., piping, equipment, and sewer lines). | • The conceptual models built-into RESRAD-ONSITE is more robust compared to DandD. RESRAD-ONSITE can handle various exposure scenarios, subsurface residual radioactivity in soils, buried radioactivity, and better assess the dose associated with smaller areas of elevated residual radioactivity. Site conditions that are incompatible with the conceptual model in RESRAD-ONSITE are listed in Table I.7. RESRAD-OFFSITE has all the capabilities of RESRAD-ONSITE (i.e., can simulate a RESRAD-ONSITE run) and can handle more complex source terms, groundwater and surface water transport. | • The conceptual model built into RESRAD-BUILD is more robust compared to DandD RESRAD-BUILD can handle multiple sources and source/receptor geometries; considers volumetric sources and shielding afforded by building materials; has a more complex building ventilation and indoor air quality model; and can better assess the dose associated with smaller areas of elevated residual radioactivity. |
| **Parameter Requirements** | • Screening Assessments—minimal parameter information is needed to use screening code (list of contaminants and concentrations). | • Site-specific physical information is needed to develop the model. Default parameters should not be used without justification (DandD default behavioral and metabolic parameters can be used with minimal justification). Default parameter distributions may be used to perform probabilistic sensitivity analyses. However, site-specific information should be used to develop or provide support for parameter distributions for risk-significant parameters if a probabilistic analysis will be used to demonstrate compliance, particularly if risk dilution is an issue (see Appendix Q for additional information on risk dilution). | • Site-specific information is needed to construct the model (building and source information). Default parameters should not be used without justification (DandD default behavioral and metabolic parameters can be used with minimal justification).<br>• Default parameter distributions may be used to perform probabilistic sensitivity analyses. However, site-specific information should be used to develop or provide support for parameter distributions for risk-significant parameters if a probabilistic analysis will be used to demonstrate compliance |

## I.6 Criteria for Selecting or Modifying Input Parameter Values

### I.6.1 Introduction

Any analytical approach to dose assessment should involve the selection of appropriate values for input parameters. Each computer modeling code or other analytical method that a licensee may use should have its own suite of input parameters. Also, unless the licensee is performing a screening analysis, each site should likely have its own defining characteristics that should be incorporated into the dose assessment through the selection of input parameter values.

This section provides general guidelines for the NRC staff to consider in evaluating a licensee's selection of values for input parameters. This section addresses three aspects of parameter value selection:

(1) selection of parameter values and distributions

(2) technical justification to support value selection

(3) evaluation of the impact of parameter selection on dose assessment results

The deterministic parameter set described in NUREG/CR–5512, Volume 1 (NRC, 1992), and implemented in DandD Version 1 has been superseded by the parameter set described in NUREG/CR–5512, Volume 3 (NRC, 1999c) and implemented in DandD Version 2. DandD Version 1 did not support probabilistic analyses and used a default deterministic input parameter set.

### I.6.2 Issues in Modifying Parameters

In addressing the three aspects of parameter value selection identified above, several issues should be discussed. First is the distinction between screening analysis and site-specific analysis, with respect to parameter value modification. Second is the appropriateness of accepting default input parameter values in site-specific analyses. Third is the level of justification expected to support the selection of site-specific input parameter values. The NRC staff should consider these issues in evaluating a licensee's dose assessment.

#### *I.6.2.1 Screening Analyses versus Site-Specific Analyses*

A licensee may perform a screening analysis to demonstrate compliance with the radiological criteria for license termination specified in 10 CFR Part 20, Subpart E. The screening analysis described in Chapter 5 of this volume requires that the licensee either (1) refer to radionuclide-specific screening values listed in the *Federal Register* (63 FR 64132 and 64 FR 68395), or (2) use the latest DandD computer code. A licensee pursuing the screening option may find that implementation of the DandD code is necessary if radionuclides not included in the *Federal Register* listings should be considered.

The NRC staff should ensure that a licensee performing a screening analysis using the DandD code limits its parameter modification to identifying radionuclides of interest and specifying the radionuclide concentrations and verify that it has not modified any other input parameter values. The output file generated by DandD identifies all parameter values that have been modified. Modifying any input parameter value from a default value will constitute a site-specific analysis.

### *I.6.2.2 Default Values versus Site-Specific Values*

DandD and many other computer codes used for dose assessment provide the user with default values for the input parameters. Often, the user need only select radionuclides to execute the code. This allows the user to quickly obtain results with very little time expended in developing input parameter sets. This is basically how DandD Version 2 was to be used for screening analyses.

While little user input or thought may be required for licensees to run common decommissioning dose modeling codes, which contain built-in default parameters, care should be taken to ensure that selected parameter values are adequately supported. Default parameter values may be inappropriate for site-specific conditions and the selected exposure scenario. When using an off-the-shelf computer code and its default parameters, the user agrees with (1) the conceptual model used by the computer code, (2) the exposure scenario, and (3) the process used to select the default parameters so that they are appropriate for the site being modeled.

Users of computer codes should have an adequate understanding of the conceptual and numerical models built into the code and how parameter inputs affect the results. Analysts should provide support for the selection of input parameters by (1) determining what characteristics of the modeled system the parameter represents (how the parameter is used in the code), and (2) developing a value for the input parameter that is appropriate for both the system being modeled and for the conceptual and numerical models implemented by the code. In fact, many default parameter values in the computer code are simply “placeholders” for site-specific data. Experience has shown that the availability of default values for input parameters can result in the user performing a “site-specific” analysis characterized by the modification of parameter values for which site data are readily available and the acceptance of default parameter values for the remaining parameters, without an adequate understanding of the implications of accepting the default parameters on the results. For site-specific analyses, the NRC staff suggests that the licensee justify using both the model and the default parameters. The level of justification needed for each parameter value is not necessarily the same. Section I.7 discusses the need to conduct uncertainty and sensitivity analyses as a method of focusing licensee and NRC staff resources on the parameters and pathways most important to dose.

The NRC staff has reviewed the DandD Version 2 and RESRAD-ONSITE, Version 6, computer codes and associated parameter distributions and considered them appropriate for dose assessments conducted to demonstrate compliance with dose-based standards in 10 CFR Part 20, Subpart E. Later versions of the code may also be appropriate for use with appropriate verification and validation, as well as QA documentation. Use of approved parameter distributions (1) promotes consistency among analyses (where appropriate), (2) helps to focus licensee and NRC staff resources on parameters considered risk-significant based on dose assessment results, and (3) facilitates the NRC’s review of the licensee’s dose assessments. Therefore, most licensees can use the code and its default parameter distributions with little justification. While default parameter distributions are approved for use in DandD and RESRAD-ONSITE, this does not mean that the default parameter distributions can be used in another computer code with no justification. Additionally, for especially risk-significant parameter values, which may be determined through probabilistic sensitivity analysis, the licenses should provide additional justification for its selection of a deterministic value for the risk-significant parameter or use of the default parameter distribution to demonstrate compliance with the dose standards in 10 CFR Part 20, Subpart E.

### *I.6.2.3 Justifying Site-Specific Parameter Values*

The NRC reviewer should evaluate whether a licensee submitting a site-specific dose assessment has demonstrated that all parameter input values are appropriate for the site being modeled. However, this does not require the licensee to submit a detailed analysis to support the values selected for each and every input parameter. Instead, the level of justification required should be based on the parameter classification and should be commensurate with the significance of the parameter relative to the dose assessment results, as evaluated through sensitivity analyses. The sensitivity analyses should reflect the relative significance of exposure pathways. Note that the relative significance of exposure pathways may change as parameters are modified.

Dose assessment input parameters may be generally classified as behavioral, metabolic, or physical. Behavioral parameters (B) collectively describe the receptor—the exposed individual for whom the dose received is being assessed. The values selected for these input parameters should depend on the behavior hypothesized for the exposed individual. Metabolic parameters (M) also describe the exposed individual but generally address involuntary characteristics of the individual. Physical parameters (P) collectively describe the physical characteristics of the site being modeled. These would include the geohydrological, geochemical, and meteorological characteristics of the site. The characteristics of atmospheric and biospheric transport up to, but not including, uptake by, or exposure of, the dose receptor would also be considered physical input parameters.

There is always uncertainty associated with the behavior of a hypothetical receptor. For this reason, the licensee may accept a generically defined receptor for its analysis. The generically defined receptor is the “average member of the critical group.” The characteristics of this exposed individual and the criteria for modifying the characteristics for a site-specific analysis are discussed in Section I.3 of this appendix. The licensee may use DandD default values for the behavioral and metabolic parameters, with limited justification, if the values are consistent with the generic definition of the average member of the critical group and if the screening group is reflective of the exposure scenario. Table I.11 lists key default behavioral and metabolic parameters used in DandD. NUREG/CR-7267 also lists key default behavioral and metabolic parameters in Tables 6-1 and 6-3; and Appendix B, Tables B-1 and B-26, for the resident farmer and building occupancy scenarios, respectively.

In site-specific analyses, all efforts should be made by the licensee to use site-specific information for important physical parameters. “Site-specific” in this context includes (1) information directly related to the site, (2) information characterizing the region that is consistent with site conditions, and (3) generic information that is consistent with the specific geohydrologic conditions at the site (e.g., consistent with the surface-soil unsaturated-zone soil classifications). The justification for site-specific physical parameter values should demonstrate that the site-specific values selected are consistent with the known or expected characteristics of the physical site being modeled. The level of justification should be based on the significance of the parameter to the results of the dose assessment. The licensee should evaluate the significance through sensitivity analyses (see Section I.7 of this appendix). Because of the importance of groundwater, the NRC staff should verify that the licensee uses site-specific values for all physical parameters (or parameter ranges) related to geohydrologic conditions. If a licensee relies on the DandD default parameter distributions for especially risk-significant parameters such as partition coefficients and soil-to-plant transfer factors, the reviewer should evaluate whether the default parameter distributions are consistent with known or expected conditions at the site to ensure that the doses are not significantly underestimated.

**Table I.11 Key DandD Deterministic Behavioral and Metabolic Parameters Mapped to RESRAD-ONSITE and -BUILD Parameters**

| DandD Parameter^ | DandD Value | RESRAD-ONSITE/BUILD Parameter | RESRAD-ONSITE/BUILD Value | Notes |
|---|---|---|---|---|
| Residential Exposure Scenario (RESRAD-ONSITE) | | | | |
| Indoor exposure period | 0.66 | Indoor time fraction | 0.5 | DandD lists as 240 days out of the year |
| Outdoor exposure period | 0.11 | **Outdoor time fraction | 0.25 | DandD lists as 40.2 days out of the year. |
| Gardening exposure period | 0.01 | | | DandD lists as 2.92 days out of the year |
| Indoor breathing rate | 0.9 $m^3$/hr | &Inhalation rate | 8400 $m^3$/y | DandD 7900 $m^3$/y |
| Outdoor breathing rate | 1.4 $m^3$/hr | | | DandD 12,000 $m^3$/y |
| Gardening breathing rate | 1.7 $m^3$/hr | | | DandD 15,000 $m^3$/y |
| Leafy vegetable consumption | 21.4 kg/y | Leafy vegetable consumption | 14 kg/y | RESRAD reduces the consumption rate based on contaminated fraction. |
| Root vegetable consumption | 44.6 kg/y | Fruit, vegetable, and grain consumption | 160 kg/y | RESRAD reduces the consumption rate based on contaminated fraction |
| Fruit consumption | 52.8 kg/y | | | RESRAD reduces the consumption rate based on contaminated fraction |
| Grain consumption | 14.4 kg/y | | | RESRAD reduces the consumption rate based on contaminated fraction |
| Beef consumption | 39.8 kg/y | Meat and poultry consumption | 63 kg/y | RESRAD reduces the consumption rate based on contaminated fraction |
| Poultry consumption | 25.3 kg/y | | | |
| Milk consumption | 233 L/y | Milk consumption | 92 L/y | RESRAD reduces the consumption rate based on contaminated fraction |
| Egg consumption | 19.1 kg/y | NA | NA | |
| Fish consumption | 20.6 kg/y | Fish consumption | 5.4 kg/y | RESRAD also includes a seafood consumption rate of 0.9 kg/y. |
| Water consumption | 1.26 L/d | Drinking water intake | 510 L/y | DandD 460 L/y |
| Indoor shielding factor | 0.55 | Shielding factor | 0.7 | This parameter may also be considered a physical parameter |

**Table I.11 Key DandD Deterministic Behavioral and Metabolic Parameters Mapped to RESRAD-ONSITE and -BUILD Parameters (cont.)**

| DandD Parameter | DandD Value | RESRAD-ONSITE/BUILD Parameter | RESRAD-ONSITE/BUILD Value | Notes |
|---|---|---|---|---|
| Building Occupancy Exposure Scenario (RESRAD BUILD) | | | | |
| Time in building | 45 hr/wk | Indoor fraction (time) | 0.5 | DandD 0.27 time fraction |
| Breathing rate | 1.4 $m^3$/hr | Breathing rate | 18 $m^3$/d | DandD 33.6 $m^3$/d |
| Ingestion rate | 1.1E-05 $m^2$/hr | Ingestion rate† | 0.0001 $m^2$/hr | |
| Loose ingestion rate | 1.1E-04 $m^2$/hr | | | |

^ NUREG/CR-5512, Volume 3 (NRC, 1999c), Table 6.87, presents a more exhaustive list of DandD parameters for the residential scenario. Average values of behavioral parameters (marked with a "B") and metabolic parameters (marked with an "M") are provided in Table 6.87.

**The outdoor and garden exposure periods must be summed to compare to the single RESRAD-ONSITE outdoor time fraction (0.12).

& The individual DandD breathing rates must be time weighted and summed to calculate an effective breathing rate of 8,600 $m^3$/hr for use in RESRAD-ONSITE.

†There is only one ingestion rate in RESRAD-BUILD. The default loose fraction in DandD is 0.1 and 0.1 x $1.1x10^{-04}$ $m^2$/hr=$1.1x10^{-05}$ $m^2$/hr.

## I.6.3 Input Parameter Data Sets

### *I.6.3.1 DandD Default Probabilistic Parameter Set*

Probabilistic analyses using the DandD computer code were performed to establish the screening values for surface-soil residual radioactivity that were published in the *Federal Register* in December 1999 (64 FR 68395). In performing these screening analyses, data were compiled for over 600 input parameters and reviewed by the NRC staff. These data are discussed in great detail in NUREG/CR–5512, Volume 3 (NRC, 1999c), and are directly incorporated into DandD Version 2. These data form the reference input parameter set for probabilistic analysis using DandD. The user is referred to NUREG/CR–5512, Volume 3, and the current version of the DandD computer code for information on the basis for the current default parameter distributions.

The DandD computer code may be used to evaluate radiological doses for two exposure scenarios: (1) the building occupancy exposure scenario and (2) the residential exposure scenario. A detailed discussion of these scenarios and the associated exposure pathways appears in NUREG-1549 (NRC, 1998d) and NUREG-CR/5512, Volume 1 (NRC, 1992).

As stated above, the licensee may use the default deterministic behavioral and metabolic parameters from NUREG/CR-5512, Volume 3 (NRC, 1999c), or the current version of the DandD computer code, with limited justification (see Table I.11). The justification should examine how the licensee's exposure scenario is consistent with the generic exposure scenario used in DandD. Similarly, a licensee may use the parameter distribution for a physical parameter, provided it justifies why the parameter distribution is consistent with the site conditions.

Note that licenses may not use deterministic physical parameter values without substantial justification (including sensitivity and uncertainty analyses).

### *I.6.3.2 DandD Default Deterministic Parameter Set*

Several default parameter sets support deterministic analyses using the DandD code. NUREG/CR-5512, Volume 1 (NRC, 1992), initially presented both the conceptual and mathematical foundation of the DandD code and the deterministic values for many input parameters. Volume 3 of NUREG/CR-5512 (NRC, 1999c) incorporated much of the parameter information from Volume 1 in developing the default probabilistic input parameter set, making corrections and updating values as necessary. Therefore, a licensee should not refer to NUREG/CR-5512, Volume 1, as a primary source for a default deterministic parameter set. Additionally, it should not use the DandD Version 1 default parameter set as a reference data set for any parameters.

Licensees may perform deterministic analyses using DandD (Version 2 or later). This would require them to change all parameter distribution types to "constant" and specify a single value for each parameter. However, the NRC staff does not intend to provide a default deterministic input parameter set to be used in conjunction with DandD. Also, a licensee intending to support decommissioning activities with deterministic dose assessments should ensure that there is sufficient information to support its demonstration of compliance (e.g., data to support parameter selection for important parameters identified through sensitivity analysis, as described in Section I.7 of this appendix).

### *I.6.3.3 RESRAD Default Probabilistic Parameter Set*

The most recent versions of the RESRAD-ONSITE, RESRAD-OFFSITE, and RESRAD–BUILD computer codes include the option to perform probabilistic dose assessments. The RESRAD team at Argonne National Laboratory worked with the NRC staff to develop default input parameter distributions that may be used to perform probabilistic dose assessments with the RESRAD-ONSITE and RESRAD–BUILD codes. These default probabilistic input parameter distributions are documented in NUREG/CR-6697 (NRC, 2000c). Guidance for performing probabilistic dose assessments with RESRAD-OFFSITE can be found in NUREG/CR-7189 (NRC, 2015b). Recently updated and newly developed RESRAD-ONSITE, RESRAD-OFFSITE, and RESRAD-BUILD parameter distributions that may be of use to licensees are available in NUREG/CR-7267.

### *I.6.3.4 RESRAD Default Deterministic Parameter Set*

As a set, the RESRAD default deterministic parameters are not considered to be acceptable for performing dose assessments in support of decommissioning. A licensee may use the default probabilistic parameter distributions described in the preceding section as a starting point for its analyses. However, the licensee should identify and justify those parameters most important to dose and provide information to support the selected values. Care should be taken to avoid use of overly broad distributions that lead to risk dilution (see Appendix Q). ANL/EAIS-8, "Data Collection Handbook to Support Modeling the Impacts of Radioactive Material in Soil" (1993), which was used to support development of parameter values and distributions in the RESRAD family of codes was updated in 2015 and re-designated as ANL/EVS/TM-14/4. ANL/EVS/TM-14/4 provides updated information on parameter definitions, typical ranges, variations, and measurement methodology. A summary of the development of the models in the RESRAD family of codes and associated parameters is found in NUREG/CR-7267 (2020a). NUREG/CR-

7267 performs probabilistic analysis using parameter values and distributions found in ANL/EVS/TM-14/4, as well as other more recent information (e.g., updated dose conversion factors), for several radionuclides. Exposure scenarios evaluated include the following: the resident farmer using RESRAD-ONSITE, building occupancy using RESRAD-BUILD, and an offsite resident scenario with water transport and offsite resident scenario with air transport using RESRAD-OFFSITE 4.0. Default behavioral and metabolic parameter values from DandD are used and key parameters and pathways are identified. The methodology and information provided in NUREG/CR-7267 may be useful to licensees performing probabilistic sensitivity analysis to identify key parameters and pathways and providing support for parameters and distributions important to dose.

### *I.6.3.5 Input Data Sets for Other Computer Codes*

A licensee may choose to use a computer code or analytical approach other than DandD or the RESRAD family of codes to perform the dose assessment in support of decommissioning. Each code or analytical approach should have a unique set of input parameters. However, there will likely be some input parameters that are also included in the DandD input parameter set.

The NRC staff should verify that a licensee lists all input parameters required in its analysis. For each parameter, the licensee should provide a discussion similar to the one in NUREG/CR-5512, Volume 3 (NRC 1999c), Chapters 5 and 6. The discussion should include the parameter name, a description of the parameter, a discussion of how the parameter is used in the dose assessment model, and the licensee's classification of the input parameter (i.e., behavioral, metabolic, or physical). For the parameters being represented by constant values, the licensee should include the range of appropriate values for the parameter, the single value selected for the parameter, and the basis for the range and selected value, including references. The level of justification to be provided in the basis should be based on the classification of the parameter (i.e., behavioral, metabolic, or physical) and the relative significance of the parameter in the dose assessment.

For input parameters classified as "behavioral" or "metabolic," the NRC staff should verify that the licensee specifies values that are consistent with the default screening values for the DandD behavioral and metabolic parameters, as long as the definition of the critical group has not been modified (see Table I.11). Consistency may depend on the conceptual and numerical models underlying the code being used and the manner in which the parameters are used in the models. Using consistent behavioral and metabolic parameter values for the default critical group may support a relatively standardized definition of the average member of the critical group among analyses. The basis the licensee provides for these parameters should identify the comparable DandD parameters and discuss any adjustments necessary to accommodate differences between DandD and the code or analytical method being used.

For the input parameters the licensee classifies as physical, with exceptions noted below, the NRC staff should verify that the licensee uses site-specific values when available. The licensee should include the soil classification for all soil units and specify consistent values for all geohydrologic parameters. For geochemical parameters, such as partition coefficients, the licensee may rely on DandD default probabilistic distributions for sensitivity analysis, as long as justification is provided to demonstrate that the ranges are consistent with geochemical conditions at the site. However, overly broad distributions for parameters that primarily affect the timing of peak dose, such as $K_d$s, can lead to risk dilution (lowering of the peak of the mean dose) as discussed in Appendix Q. Licensees and reviewers should evaluate the potential for

risk dilution if the peak of the mean dose is used as the metric to demonstrate compliance with radiological criteria for license termination in probabilistic analyses. The licensee may need to modify the default parameter distributions to ensure consistency with site conditions. Additionally, it is important to note that the distributions may not be applicable to codes other than DandD. For meteorological parameters, the licensee should use values that are based on applicable site or regional data. For physical parameters related to atmospheric and biospheric transport, the licensee may accept DandD default parameter ranges with minimal justification for sensitivity analysis, using NUREG/CR-5512, Volume 3 (NRC, 1999c), as a starting reference point. Physical parameters related to biosphere transport would include parameters such as crop yields, animal ingestion rates, transfer factors, and crop growing times. The NRC staff should evaluate whether the justification provided by the licensee demonstrates that the default values are consistent with conditions at the site. Generally, $K_d$s and transfer factors are particularly risk significant. If the sensitivity analysis reveals the importance of these parameter values, care should be taken to select a deterministic value that is supported by site-specific data or to ensure that parameter distributions used in probabilistic analyses to demonstrate compliance with the performance objective are not expected to underestimate the potential dose.

#### *I.6.3.6 Internal and Direct Exposure Dose Conversion Factors*

The NRC staff should review the dose conversion factors for inhalation and ingestion, to ensure that the factors used are those developed by EPA, published in Federal Guidance Report No. 11 (EPA, 1988). Similarly, the review should ensure that EPA's external dose factors, although they may correct for actual area, published in Federal Guidance Report No. 12 (EPA, 1993), are used or another appropriate code such as Microshield is used. These dose factors were selected to ensure that the dosimetry models used in deriving these factors are consistent with NRC regulations in 10 CFR Part 20.

More recent dosimetry methods may be acceptable for use and will be evaluated on a case-by-case basis. For example, licensees may request an exemption from 10 CFR Part 20 to use more recent dose conversion factors (e.g., ICRP 72). ICRP 72 provides age dependent committed effective dose conversion factors for ingestion and inhalation of radioactivity. If a licensee proposes to use ICRP 72, the selection of exposure scenarios and critical groups should take into account differences in potential exposure due to age. Licensees should avoid "picking and choosing" dosimetry methods for different radionuclides (e.g., Federal Guidance Report No. 11 for six radionuclides and current international dose conversion factors for three radionuclides).

### I.6.4 Recommended Approach to Parameter Modification

Any analysis that does not meet the conditions of a screening analysis may be considered a site-specific analysis. This will include all analyses using the DandD computer code, where one or more input parameter values have been modified from default ranges (or values for behavioral and metabolic parameters), as well as analyses using analytical methods or computer codes other than DandD.

#### *I.6.4.1 Modifying the DandD Default Probabilistic Parameter Set*

A reviewer should expect that a licensee who is modifying parameter values for a site-specific analysis using DandD is cognizant of the following:

(1) what the parameter represents

(2) how the parameter is used in the DandD code

(3) the basis for the default parameter value

(4) which parameters are physically or numerically correlated

NUREG/CR-5512, Volumes 1–3, describes in detail what each parameter is intended to represent. Volume 1 (NRC, 1992) provides the original parameter definitions but has been superseded by Volume 3 (NRC, 1999c) for parameter values. Volume 1 also provides the mathematical formulations underlying the DandD code that should allow the user to (1) understand how each parameter is used and the implication of parameter modification on the resulting calculated dose, and (2) identify numerical correlations among parameters. Volume 2 (the DandD user's manual) redefines several of the input parameters and mathematical formulations based on implementing the Volume 1 methodology in the DandD computer code. Finally, Volume 3 contains a detailed discussion of most input parameters, allowing the user to fully understand the basis for the default ranges. Volume 3 includes a parameter description and a discussion of how parameters are used in the code, a review of the information sources on which the default values are based, a discussion of uncertainty in the default parameter values, and insight into the selection of alternative parameter values. The DandD user performing site-specific analyses should be cognizant of the information provided in the three volumes of NUREG/CR-5512.

A licensee may modify DandD behavioral (B) and metabolic (M) input parameter values for the building occupancy and residential exposure scenarios to reflect the characteristics of the average member of a *site-specific* critical group. NUREG/CR-5512, Volume 3, provides the basis for the default value for each behavioral and metabolic parameter (see Table I.11). If the licensee modifies the values for these parameters, the NRC staff should verify that the licensee has defined a *site-specific* critical group. The licensee may provide site-specific parameter distributions that reflect the variability of the behavior of the average member of the site-specific critical group, or the licensee may use the mean of the site-specific information as a constant-value input for these parameters, consistent with the concept of the "average member" of the critical group. The level of justification required to support modification of behavioral and metabolic parameter values should be consistent with the sensitivity of results to changes in the parameter value.

For the DandD building occupancy exposure scenario, there are only three physical parameters: the resuspension factor (Rfo*), which is derived from the loose fraction (Fl) and the loose resuspension factor (Rfo). If the inhalation pathway dose can be significant, these parameters should be carefully selected. The default removable fraction assumed in DandD is 10 percent. If the removable fraction is expected to be greater than 10 percent, the licensee should account for higher removable fractions that might increase the resuspension factor. Because the default removable fraction is multiplied by the loose resuspension factor to derive the resuspension factor in DandD, either the removable fraction or the resuspension factor can be adjusted to account for removable fractions greater than the default value of 10 percent. If site conditions are consistent with assumptions in NUREG-1720 (with respect to activities and exposure scenarios, ventilation conditions, and low removable fractions at the time of decommissioning), NUREG-1720 (NRC, 2002) recommended the use of resuspension factor value or parameter distribution with minimal justification.

There are many more physical parameters for the DandD residential exposure scenario. The physical parameters may be considered in several groups. The following physical parameters address the geohydrologic conditions:

- Unsaturated Zone Thickness (H2)
- Soil Classification (SCSST)
- Porosity Probability (NDEV)
- Permeability Probability (KSDEV)
- Parameter “b” Probability (BDEV)
- Water Application Rate (AP)
- Surface Soil Porosity (N1)
- Unsaturated Zone Porosity (N2)
- Surface Soil Saturation (F1)
- Unsaturated Zone Saturation (F2)
- Infiltration Rate (INFIL)
- Surface Soil Density (RHO1)
- Unsaturated Zone Density (RHO2)
- Surface Soil Permeability (Ksat1)
- Soil Moisture Content (sh)

For these physical parameters, the licensee should use site-specific distributions and values. (As stated previously, “site-specific” in this context includes (1) information directly related to the site, (2) information characterizing the region that is consistent with site conditions, and (3) generic information that is consistent with the specific geohydrologic conditions at the site (e.g., consistent with the unsaturated zone soil classification)).

The NRC staff should verify that the licensee has provided site-specific information for the thickness of the unsaturated zone and the soil classification. In addition, the licensee should ensure that the water application rate is consistent with the irrigation rate (behavioral parameter), if the licensee modifies the irrigation rate. Alternatively, the licensee may demonstrate, through sensitivity analyses, that the dose assessment results are insensitive to these parameters and use the default ranges.

Values for the derived parameters will be generated internally according to the soil classification indicated and the uniform distributions defined for the porosity probability (NDEV), the permeability probability (KSDEV), and the parameter “b” probability (BDEV). The NRC staff should verify that the licensee has not modified the uniform distributions for these three

parameters. If site-specific data are available, the licensee may proceed to modify the derived geohydrologic parameters, consistent with the information presented in NUREG/CR-5512, Volume 3 (NRC, 1999c).

The only geochemical parameter used in DandD is the element-specific partition coefficient. As documented in NUREG/CR-5512, Volume 3 (NRC, 1999c), the partition coefficients at a site are generally both dependent on geochemical conditions and independent of soil classification. If the licensee has used the default distributions, the reviewer should evaluate whether the defaults are consistent with known or expected conditions at the site.

The following physical parameters address radionuclide transport through the atmosphere and exposure to direct radiation:

- outdoor shielding factor (SFO)
- flood dust loading (PD)
- indoor resuspension factor (RFR)
- outdoor dust loading (CDO)
- indoor dust loading (CDI)
- indoor/outdoor penetration factor (PF)
- gardening dust loading (CDG)

The remaining physical parameters address characteristics of transport through the biosphere:

- growing periods (produce, forage, grain, hay) (TG_(#))
- animal product specific activity (SATac)
- livestock feeding periods (TF_(#))
- animal product yields (YA(#))
- interception fractions (R_(#))
- translocation factors (T_(#))
- contaminated fractions (x_(#))
- crop yields (Y_(#))
- wet-to-dry conversion factors (W_(#))
- animal ingestion rates (Q_(#))
- mass-loading factors (ML_(#))
- carbon fractions (fc_(#))

- hydrogen fractions (fh_(#))
- hydrogen fraction: soil (fhd016)
- tritium equivalence: plant/soil (sasvh)
- tritium equivalence: plant/water (sawvh)
- tritium equivalence: animal products (satah)

These two groups of physical parameters describe characteristics of the transport of radionuclides through the atmosphere or biosphere up to the point of ingestion or inhalation by, or external exposure to, the receptor. The licensee may accept the default distributions for these parameters as long as they are consistent with conditions that may exist at the site in the future. The licensee should review the basis given in NUREG/CR-5512, Volume 3 (NRC, 1999c), for the default distributions, to determine whether the basis is consistent with conditions hypothesized for the site. If not, the licensee should modify the input values accordingly. The NRC staff should ensure that the licensee documents this assessment for each of the physical parameters. Note that modifying several of these parameters (e.g., crop yields, animal product yields) should affect the derived behavioral parameters (e.g., area of land cultivated).

For the physical parameters, the licensee may use representative distributions or values. A representative distribution should take into account spatial and temporal variation of the parameter at the site. A representative distribution, for example, would be a precipitation rate based on the historical precipitation data for the site, if available, or from surrounding defensibly relevant monitoring locations. The arithmetic or geometric mean value is often used in defining a representative value. However, the calculation of a mean value should be weighted to account for nonuniform sampling or other nonuniform parameters (e.g., material volume) and parameter sensitivity and uncertainty. The licensee is not required to routinely adopt worst-case, bounding, upper- or lower-percentile, or other overly conservative values in defining distributions.

The review of this information should be facilitated if the licensee presents the information in a tabular or list format. The NRC staff should verify that the licensee has listed every DandD input parameter with the default screening distributions or value (for behavioral or metabolic parameters). For those parameters for which the licensee is using site-specific values (e.g., the physical parameters), the licensee should provide the range of plausible values for the site, the selected distribution or value, and supporting justification, including references.

#### *I.6.4.2 Modifying the RESRAD Default Probabilistic Parameter Set*

A licensee using the RESRAD-ONSITE, RESRAD-OFFSITE, or RESRAD–BUILD codes may change parameters from the default values to reflect a site-specific critical group or site-specific conditions, or to incorporate site-specific data. As discussed in the preceding section, the NRC staff should expect that a licensee who is modifying parameter values for a site-specific analysis is cognizant of the following:

- what the parameter represents
- how the parameter is used in the code

- the basis for the default parameter value
- which parameters are physically or numerically correlated

The licensee should refer to the current code documentation to determine the basis for the parameter distributions and how they are used in the code and provide references to the documentation. With respect to the basis for the default parameter distributions and values, the licensee should refer to NUREG/CR-6697 (NRC, 2000c) and NUREG/CR-7267 (NRC, 2020a).

When modifying parameter distributions and values, the licensee should consider whether the parameters are classified as behavioral, metabolic, or physical. For behavioral and metabolic parameters for which probability distributions have been developed, the licensee may adopt the DandD default distribution, or the mean of the DandD default distribution, as long as the licensee has not modified the definition of the critical group (see Table I.11). For behavioral and metabolic parameters for which distributions have not been developed, the licensee should use values or distributions that are consistent with the DandD default distributions, as applicable.

A licensee may modify behavioral and metabolic default input parameter values to reflect the characteristics of the average member of a *site-specific* critical group. The licensee may modify the values for these parameters if the licensee has defined a *site-specific* critical group. The licensee may provide site-specific parameter distributions that reflect the variability of the behavior of the average member of the site-specific critical group use the mean of the site-specific information as a constant-value input for these parameters, consistent with the concept of the “average member” of the critical group. The level of justification required to support modification of behavioral and metabolic parameter values should be consistent with the sensitivity of the results to changes in the parameter.

For the physical parameters, the licensee should use site-specific information addressing geohydrologic and meteorologic conditions. The level of justification for the parameter values should be based on sensitivity analyses. Alternatively, sensitivity analyses may be used to support the use of default distributions or representative values. Care should be taken to ensure that use of overly broad distributions for parameters that effect the timing of peak dose do not lead to risk dilution (see Appendix Q for additional details).

For the physical parameters describing geochemical conditions (i.e., $K_d$s), the licensee should use values that are consistent with known or expected site conditions. The RESRAD default distributions may be acceptable for sensitivity analyses, as long as the values are consistent with site-specific conditions. If the peak of the mean from a probabilistic analysis is used to demonstrate compliance with radiological criteria for license termination, as stated in the previous paragraph, the range of $K_d$ values should not be overly broad (i.e., significantly outside of expected site-specific conditions) due to the potential for risk dilution, or the spreading out of peak dose over time leading to an artificially low peak of the mean dose that is not reflective of site risk. A sensitivity analysis can be used to determine the relative importance of the parameter values on dose. Parameter distributions for parameters having a significant impact on the results should be carefully selected based on site-specific conditions.

For the remaining physical parameters (atmospheric and biospheric transport), the licensee may use distributions or representative values that are consistent with the RESRAD default distributions, as applicable, as long as the default distributions are consistent with known or expected site conditions.

### *I.6.4.3 Sensitivity Analyses*

The level of justification required to support site-specific parameter values should be commensurate with the sensitivity of the results of the dose assessment to the selected values. Section I.7 of this appendix discusses sensitivity analyses in detail.

### *I.6.4.4 Site-Specific Distribution Coefficients for Soil or Concrete*

The following describes an acceptable approach for the developing input $K_d$ values for soil or concrete for use in site-specific dose modeling codes.

It is noted that $K_d$ values commonly reported in the literature may vary by as much as six orders of magnitude for a specific radionuclide. Generally, no single set of ancillary parameters, such as pH and soil texture, is universally appropriate in all cases for determining appropriate $K_d$ values. Although $K_d$ values are intended to represent adsorption, they are in most cases a lumped parameter representing a myriad of processes. Given the above, the proper selection of a range of $K_d$ values, for either soils or concrete, from the literature will require judicious selection.

The licensee is encouraged to use sensitivity analyses to identify the importance of the $K_d$ parameter on the resulting dose. The sensitivity analysis should encompass an appropriate range of $K_d$ values. The input range for the sensitivity analysis may be obtained from literature, DandD default distributions, RESRAD probabilistic default distributions, and ideally site-specific information.

A commonly used reference for $K_d$ data, Sheppard and Thibault, "Default Soil Solid/Liquid Partition Coefficients**,** $K_d$**s,** for Four Major Soil Types: A Compendium" (Sheppard and Thibault, 1990), provides lognormal distribution parameters for a long list of radionuclides by soil type. Sheppard et al. "Soil Nuclide Distribution Coefficients and Their Statistical Distribution" (Sheppard et al., 1984), and Sheppard and Evenden, "Comparison of Partition Coefficients for $^{54}$Mn and Soil-Extractable Mn, Including Relationship to Plant Uptake" (Sheppard and Evenden, 1989) are cited in Sheppard and Thibault (1990) as providing support for the use of lognormal distributions to characterize $K_d$s. Although lognormal distributions are commonly used, it is important to note that the selection of an appropriate parameter distribution to describe $K_d$ uncertainty or variability should be based on the goodness of fit of the model to the data available to define the distribution. Undoubtedly, Sheppard and Thibault's (1990) conclusion that $K_d$ data follow a lognormal distribution is, based in large part, on the inclusion of a wide range of values from the literature representative of a number of sites. Depending on the objective of the modeling exercise (e.g., to understand sensitivity of the results to potential range of $K_d$s representative of a number of conditions and sites, or to understand uncertainty in the results given the potential variability of $K_d$s for a particular site when uncertainty has already been reduced with respect to the site characteristics), the most appropriate parameter distribution to describe the $K_d$ or coefficients for a particular radionuclide and site may vary. Therefore, the most appropriate distribution for any particular site, or project, cannot be determined *a priori* and should be based on a careful review of the data. The licensee can use codes such as MATLAB® statistics toolbox to develop parameter distributions and evaluate parameter correlations, if sufficient data are available.

Using the results of the sensitivity analysis, the licensee can obtain support for its selection of $K_d$ value. For example, if higher $K_d$ values result in a larger dose, an input $K_d$ value should be selected from the upper range of values that represent site-specific conditions. If lower $K_d$

values result in the larger dose, an input $K_d$ value should be selected from the lower range of values that represent sites-specific conditions. If dose and compliance risk are sensitive to the selection of $K_d$, it may be necessary to conduct experiments using site materials to provide support for $K_d$ values used in dose modeling. For those isotopes where the $K_d$ does not have a significant impact on the dose assessment based on a sensitivity analysis (i.e., the dose results are not sensitive to $K_d$), limited justification will be needed to support selection of the parameter value.

As a result of the NRC staff review of dose modeling to support decommissioning and performance assessment modeling for low-level waste disposal, a number of observations and recommendations are noted with respect to the use of $K_d$s in dose modeling to demonstrate compliance with radiological criteria for license termination or in performance assessment analyses. Table I.12 lists key considerations related to selection of parameter values and consideration of uncertainty.

**DANDD**

Data in Table 6.86 of NUREG/CR-5512, Volume 3, "Residual Radioactive Contamination from Decommissioning—Parameter Analysis," issued 1999 (NRC, 1999c), may be useful in the development of $K_d$ parameter distributions in a sensitivity analysis but should be representative of or expected to encompass site-specific conditions.

**RESRAD FAMILY OF CODES**

Licensees should not use RESRAD default parameter values (including $K_d$ values). The code-included defaults serve primarily as place holders that enable it to be run; it was assumed that site-specific values would be developed. However, the default $K_d$ parameter distributions, developed for RESRAD-ONSITE, Version 6.0, may be useful in the development of $K_d$ parameter distributions for use in sensitivity analysis but should be representative or expected to encompass site-specific conditions. Updated information on parameter distributions is provided in NUREG/CR-7267.

Following sensitivity analysis with the appropriate $K_d$ ranges, support can be provided for a value in the upper or lower range of the distribution, whichever endpoint results in the highest derived dose. If the use of $K_d$ values near the endpoints of the distribution leads to doses above the release criteria, a site-specific $K_d$ value may be needed to demonstrate compliance with the radiological criteria for license termination.

**METHODS FOR DEVELOPMENT OF SITE-SPECIFIC $K_d$**

There are five general methods used to measure $K_d$ values: laboratory batch method, *in situ* batch method, laboratory flow-through (or column) method, field modeling method, and $K_{oc}$ method (EPA, 1999, page 62). Each approach has its own built-in assumptions and consequently, its own distinct advantages and disadvantages compared to other methods. Ultimately, differences in assumptions can lead to significant differences in calculated $K_d$s when using these various methods as discussed in more detail below.

Experimental methods used to calculate $K_d$s have several built-in assumptions and limitations. For example, intrinsic to the batch method is the assumption of achieving equilibrium (or at least constant concentration of the contaminant) and that this equilibrium is entirely reversible. This

can be addressed by taking measurements to ascertain that concentrations have become constant. Additionally, results of an experiment alone cannot distinguish between adsorbed and precipitated species with precipitation sometimes being attributed to sorption.

**Table I.12 Distribution Coefficient Parameter and Uncertainty Evaluation Considerations**

| Category | Consideration or Recommendation |
|---|---|
| **Parameters** | Literature values based on soil type are not considered site-specific values. Furthermore, $K_d$s may not be strongly correlated to soil type. |
| | Development of site-specific $K_d$s (versus use of literature values) may be necessary to inform the decision-making process, if the decision being made is sensitive to selection of the parameter value over a plausible range of parameter space. |
| | Experiments used to develop site-specific parameters should be representative of field conditions. Complicating mechanisms that may be present in the field may be difficult to reproduce, study, or interpret in the laboratory. Experimental artifacts may lead to erroneous conclusions on sorption potential in the field. Complicating mechanisms that may affect sorption include variable saturation, presence of a colloidal or anion exclusion mechanism, particle size heterogeneity, variable solution chemistry, and differences in sorption and desorption reactions and rates. Experimental artifacts include issues with material storage and preparation, sorption on filters or containers, nonrepresentative soils or aqueous chemistry, introduction of incompatible components that may chemically alter the system, and flow channeling or variability. |
| **Treatment of Uncertainty** | Overly broad $K_d$ parameter distributions in probabilistic analyses could lead to risk dilution and poor decision-making. A significant difference between peak of the mean dose versus mean of the peak dose is an indication of risk dilution. If there is evidence of risk dilution, effort should be made to constrain the $K_d$ parameter range. A metric different than the peak of the mean dose could also be used to evaluate compliance. Broad distributions may be acceptable for the purpose of conducting a sensitivity analysis. |
| | In some cases, when ground-water-dependent pathways dominate the dose, lower distribution coefficients may lead to higher doses. In other cases, when non-ground-water-dependent pathways dominate the dose, higher distribution coefficients may lead to higher doses, particularly if in-growth of daughter products can increase the dose/risk over time. |

Typically, only the total aqueous concentration of a contaminant is measured, with no analysis of speciation; implicit in this measurement is the assumption that only one aqueous phase species is present. A potential problem with assuming that only one aqueous species is present when, in reality, there are multiple species of varying mobility, is that the measured $K_d$ represents an average $K_d$ that may not provide sufficient information to accurately model contaminant transport. For example, if two species are present, one with significantly lower $K_d$ than the other, faster travel times and potentially higher peak concentrations associated with the

more mobile fraction will not be simulated using an average $K_d$, potentially leading to an underestimate of contaminant concentrations. Column experiments have distinct advantages over batch experiments in their ability to identify fractions of significantly higher mobility, as evidenced by significantly earlier breakthrough times for a fraction of the total source activity. Although column experiments may be able to identify relatively mobile fractions, accurately identifying mechanisms and selecting appropriate transport parameters for predictive modeling may be more difficult.

Although various experimental methods are routinely used to measure or calculate distribution coefficients, experimental conditions can compromise their effectiveness in developing representative distribution coefficients. Experimental conditions that can affect $K_d$ measurements include pH; redox speciation; dissolved gas and solute concentrations; temperature and pressure; physical, chemical and mineralogical characteristics of available sorbing surfaces; stirring rates and/or flow rates (P.G. Glynn, "Modeling Np and Pu Transport with a Surface Complexation Model and Spatially Variant Sorption Capacities: Implications for Reactive Transport Modeling and Performance Assessments of Nuclear Waste Disposal Sites" (Glynn, 2002)), among others. As a result, thought and care should be applied to developing the experimental plan for sorption studies. The selection of materials, including appropriate site-specific sediment and groundwater, is of primary importance. American Society for Testing and Materials (ASTM) C-1733, "Standard Test Method for Distribution Coefficients of Inorganic Species by the Batch Method" (ASTM, 2010) discusses the necessary precautions that have been found to be important.

There are a variety of published tests for $K_d$ measurements. Experimenters should be judicious in their selection of methods to develop sorption parameters. Some tests are appropriate to measure sorption of metals and radionuclides, while some are designed for organics. Some tests use batch, equilibrium methods, while others are flow through. Some tests address sorption while others look at desorption (through the use of leaching tests). The following section briefly discusses some of these tests and their advantages and disadvantages. EPA (1999) provides a good evaluation of the different methods used to derive distribution coefficients and the advantages, disadvantages, and assumptions for each of the methods.

*Batch Methods to Determine $K_d$s*

Batch adsorption tests for the distribution coefficient (or partition coefficient, or $K_d$) are essentially simple tests in which a known quantity of a solid (e.g., soil or mineral sample) is contacted with a volume of water containing a known (often spiked) concentration of some species of interest. After allowing the system to come to steady state (actually, it is constant concentration), the aqueous concentration of the species of interest is measured and compared to the starting concentration. Typically, just the solution is measured and the mass of sorbed species is determined by subtraction of the before and after concentrations. While in principle this is a simple test, details are important and can make big differences in the $K_d$ value obtained. Among the important factors, probably the greatest is the pH of the contact solution. For the NRC's purposes, it is important that site-specific materials, both soil and groundwater, be used and that these materials be handled in ways that do not disturb their chemical properties.

Currently, ASTM has two approved batch $K_d$ methods. There were several earlier ASTM tests that have either been withdrawn or have not been renewed (ASTM tests need to be renewed every 5 years or they are dropped from the published methods). Both are batch tests and both are specifically designed to quantify the sorption of inorganic species. This is an important point; tests for organics can be more complex than those for inorganics because, for organics,

one needs to consider degradation and volatilization of the contaminant. A few tests for organic contaminants are listed below but are not described.

ASTM D4646-3 (2016), *Standard Test Method for 24-h Batch Type Measurement of Contaminant Sorption by Soils and Sediment*, is "applicable in screening and providing relative rankings of a large number of samples for their sorption affinity in aqueous leachate/geomedia suspensions." While this method may be useful for screening, it is generally considered to be too short to establish steady-state conditions, which are needed for models. This time dependence is very much controlled by the element or species being tested; for example, anionic species typically need a number of days to reach constant concentration. The method suggests using ASTM D-4319 (2002) for longer tests. However, D-4319 has not been renewed (the standard was withdrawn in 2007). Because of the short duration of the test, D-4646 is not recommended for generating $K_d$s for use in performance assessment modeling.

ASTM C-1733-10 is a new method, the scope of which is to "quantify uptake onto solid materials by a batch method." It is intended "to assess sorption of dissolved ionic species subject to migration through pores and interstices of site-specific geomedia." This method provides a substantial discussion of factors that can influence $K_d$ values, with emphasis on being aware of these effects or controlling them in the laboratory. The method discusses its use with radioactive tracers and recommends a preliminary $K_d$ test to determine the time needed to reach constant solution concentrations of the species of interest. A correction method is given for materials containing grains larger than 2 mm. The "Significance and Use" section of the method states, "Because of the sensitivity of $K_d$ to site-specific conditions and materials, the use of literature derived $K_d$ values is strongly discouraged." This method is suggested in Regulatory Guide 4.7, Revision 3, "General Site Suitability Criteria for Nuclear Power Stations," issued March 2014 (NRC, 2014), that was issued as DG-4021 in December 2011. Ebert and Petri, "Uptake of Cs and Sr on San Joaquin Soil Measured Following ASTM Method C-1733," (Ebert and Petri, 2012) evaluated the repeatability and sensitivity to test parameters for this test method; for example, an average $K_d$ of 373 mL/g for Cs was determined with a standard deviation of 21 mL/g or 5.6 percent.

Other batch methods are published but not as official consensus standard methods. One is the "Standard Method Used at Pacific Northwest National Laboratory for Measuring Laboratory Batch Kd Values." It is published as Appendix C in Volume 1 of EPA 402-R-99-004A, "Understanding Variation in Partition Coefficient, Kd, Values Volume 1: The $K_d$ Model of Measurement, and Application of Chemical Reaction Codes" (EPA, 1999). This method is oriented toward radioactive tracers and stipulates a 7-day contact period unless time is a parameter to be investigated.

EPA published another batch method in 1992 in a Technical Resource Document entitled "Batch-Type Procedures for Estimating Soil Adsorption of Chemicals," EPA/530/SW-87/006-F (Roy et al., 1992). Chapter 17 of this informative report, *Laboratory Procedures for Generating Adsorption Data,* contains a detailed procedure with provisions for both organic and inorganic contaminants. In this method, emphasis is placed on assessing the impacts of various test parameters such as contact time and solid/solution ratios. The results are used to generate isotherms.

Serne and Relyea, in "Waste/Rock Interactions Technology Program: The Status of Radionuclide Sorption-Desorption Studies Performed by the WRIT Program," issued April 1992 (PNL-3997), discuss laboratory methods development for batch tests and report on a round-robin test that was conducted among nine participants using Cs, Sr, and plutonium (Pu) tracers.

Results were very scattered and Pu adsorption data were regarded as not satisfactory, with two to three orders of magnitude differences. On examining the results of the tests, the problem was ascribed to Pu retention on containers and formation of Pu particles. Overall, it was concluded that several uncontrolled parameters may have affected results: (1) method of tracer addition to solution, (2) solution-to-rock ratio, (3) initial tracer concentration in influent solution, (4) particle size distribution, (5) solid-solution separation method, (6) sample containers, and (7) temperature.

Although they are not the subject of this report, a few standard test methods for adsorption of organic chemicals are listed below. While they are oriented toward organics, these methods present useful information for all sorption tests.

- ASTM E1195-01(2008), "Standard Test Method for Determining a Sorption Constant ($K_{oc}$) for an Organic Chemical in Soil and Sediments"
- EPA 712-C-08-009, (2008), "Fate, Transport and Transformation Test Guidelines, OPPTS 835.1230, Adsorption/desorption (Batch Equilibrium)"
- OECD 2000, "OECD Guideline for Testing of Chemicals, Adsorption-Desorption Using a Batch Equilibrium Method, Method 106"

*In Situ Batch $K_d$*

*In situ* batch methods may also be used to develop site-specific $K_d$s. Using this method, cored samples are extracted directly from the aquifer. Phase separation is accomplished through centrifugation or filtration and contaminant concentrations in both phases are measured. The advantage of this approach, compared to the laboratory $K_d$ method, is that actual samples are used to calculate the $K_d$s. There is also a greater likelihood that the sample is closer to equilibrium. A disadvantage of this approach is that surface concentrations of most metal contaminants are typically low and difficult to measure. Energy dispersive x-ray analysis and x-ray fluorescence, with typical detection limits around 10,000 and 100 ppm, respectively, are commonly used to measure contaminant concentrations on the solid phase. Lower detection limits may be obtained by dissolving the solid phase sample with acids and through use of inductively coupled plasma spectroscopy, inductively coupled plasma/mass spectroscopy, or atomic adsorption spectroscopy techniques. In addition to the detection limit problem and similar to problems with the laboratory batch method, no differentiation can be made between sorption and precipitation. Additionally, some trace contaminants may be associated with crystalline lattice sites of minerals present in soils rather than being reversibly sorbed to mineral surfaces. For anthropogenic radionuclides present at trace levels, it may be acceptable to assume that the total mass measured on the solid is reversibly sorbed and that precipitation and lattice site associations are negligible (EPA, 1999).

*Column Experiments to Determine $K_d$s*

Column tests are used much less often than batch tests and, in fact, searching the literature has resulted in no standard methods for soil adsorption tests using columns. Of course, all sorts of adsorption columns are used, from laboratory scale to immense industrial columns, so the concept is readily and often applied to soils. There are papers and reports that describe the apparatus and methods specifically for soil column tests and a few that compare results of batch and column tests. Most column tests are used to assess the behavior of organic chemicals, but there are many that investigate inorganic chemicals. There are standard column leach test

methods that have many similarities to adsorption experiments, especially the apparatus used; see ASTM D-4874; *Leaching Solid Material in a Column Apparatus*.

Column adsorption tests are substantially more complex, both physically and analytically, than batch tests. Relyea et al. ("Methods for Determining Radionuclide Retardation Factors: Status Report," 1980 (Relyea et al., 1980)) provide a good comparison and discussion of the two types of tests. The experimental apparatus for column tests includes a column that will contain the material to be tested and not leak, a low-flow pump, a reservoir to hold the influent solution, and a sample collection system. Compared to a batch test, more information is needed to interpret a column test, including column dimensions, mass of added material, porosity, bulk density, flow rate, and sample collection times. Typically, column tests require many samples over time, as opposed to a limited number in batch tests (to establish constant concentration). Depending on the $K_d$ of the system being tested, column tests may need to run for weeks or months (or more) to establish breakthrough. Alternatively, the column material can be extruded and sectioned and the solids analyzed for profiles of the species of interest. Interpretation of results generally requires computer models, and it is good practice to perform a measurement of flow rate and dispersion using a nonreactive tracer such as tritiated water.

There is general agreement on some experimental procedures for column tests. Columns should be run with water entering from the bottom of the column. This "upflow" mode helps eliminate preferential pathways. Columns should be more than 20 times the width of the largest grains of the tested material. Column fabrication materials should be chosen to minimize sorption on walls, for example. The design should include a means of holding the tested material in place. Flow rates should be considered carefully, because they control residence time. Serne and Relyea (1982) discuss some advantages and disadvantages of column methods. Relyea et al (1980) provide, in Appendix C, a method for a column procedure and data reporting. Little information is given for the apparatus; the emphasis is on data reporting and interpretation.

Fjeld et al., "Final Report Column Tests to Study the Transport of Plutonium and Other Radionuclides in Sedimentary Interbed at INEEL" (Fjeld et al., 2000), provide some methodology for column tests used to investigate transport of radionuclides such as Sr, Am, Pu, neptunium (Np), and U. The analysis of column data, the interpretation of breakthrough curves, and the geochemistry controlling them are useful. Although they are focused on organics, column packing, conservative tracer tests, and retention time concerns are universal.

*Field Transport Rate Approach*

This field approach to deriving $K_d$ values uses groundwater monitoring data, source term data, and other flow and transport parameters to back-calculate $K_d$s based on travel times. A dual tracer test, using a nonsorbing conservative tracer and a tracer of the element of interest may be useful. Analytical models can be used to solve for the retardation factor from which a distribution coefficient can be calculated. Alternatively, numerical methods can be used to determine $K_d$ through calibration with measured plume concentrations.

*Potential Problems for Sorption Experiments*

The following sections describe several experimental artifacts and methods of overcoming them to make them more useful for predicting contaminant mobility in the environment.

Hysteresis—One complicating factor in sorption experiments is that many contaminants have been observed to sorb more readily than desorb from mineral or organic surfaces, a phenomenon referred to as hysteresis. Most batch experiments measure sorption, not desorption, although desorption may be much slower than sorption. If sorption reactions are not completely reversible or desorption is kinetically limited, the contaminant may be present for longer periods of time than assumed and remedial options (e.g., pump and treat) may be hampered.

Gravel Issue—The "gravel issue" is the problem that transport modelers face when converting laboratory-derived $K_d$ values based on experiments using the less-than-2-mm fraction into values that can be used in systems containing particles greater than 2 mm in size. For example, experiments using less than a 2-mm fraction may overestimate sorption due to the higher surface area of smaller particles. EPA (1999) proposed two methods: the first considers the fraction of larger and smaller size particles present at the site. The experimentally derived $K_d$ using 2-mm-size particles is adjusted to account for larger materials. To adjust the $K_d$, site soils are characterized to identify the fraction of particles with diameters greater than 2 mm and a $K_d$ of 0 L/kg is applied to the larger particle size fraction. This approach has been adopted by ASTM C-1733. The second approach adjusts the experimentally derived $K_d$ using the ratio of surface area in the experiment to the surface area of the material in the field. This method assumes a relatively small specific surface area of gravel-size material.

Colloid Formation—While mathematical models are available to consider a separate colloidal phase, evaluating the impact of colloids on contaminant transport predictions is hampered by the difficulty in characterizing the colloidal phase in the field and through experimentation. Additionally, a lack of mechanistic information on the factors that lead to the generation and removal of colloids in site-specific applications adds to the uncertainty in contaminant transport projections. Contaminant mass sorbed to colloidal particles that remain in suspension after separation of the solid and aqueous phases is sometimes included with the total aqueous phase concentration, resulting in a significant underestimation of the distribution coefficient. For example, use of a centrifuge for liquid/solid phase separation may result in formation of a very thin zone of suspended particles at the liquid surface where surface tension holds finer particles in solution. When pipets are used to remove supernatant solution, the pipet should be inserted sufficiently below the liquid surface to limit extraction of suspended particles. Likewise, the pipet should be significantly above the bottom of the sample to avoid extraction of settled particles. Filtering following centrifuging is desirable; however, care should be taken to avoid filters that might sorb contaminants. Pretreatment of filters may also help minimize container sorption biases.

Materials—The selection of inappropriate materials for the experiments can seriously degrade the results of sorption tests. The soil or other solid phase must be representative of the material that the contaminant will flow through. Similarly, the aqueous phase must be sampled from the appropriate unit or must be carefully matched to the subject water, if simulated groundwater is used. Most especially, speciation altering parameters (such as pH and complexing ions) must be controlled. Care must be given to storage of these materials before the experiments start. Experimental materials should be judiciously chosen, making sure that no sorption takes place on containers or filters.

## I.7 Uncertainty/Sensitivity Analyses

### I.7.1 Introduction

Uncertainty is inherent in all dose assessment calculations and should be considered in regulatory decision-making. In general, a dose assessment has three primary sources of uncertainty: (1) uncertainty in the models, (2) uncertainty in the scenarios, and (3) uncertainty in the parameters (NUREG/CR-4604, "Statistical Methods for Nuclear Material Management," issued December 1988 (NRC, 1988a); DOE Conference, "Treatment of Uncertainty in Low-Level Waste Performance Assessment," held November 19–21, 1991 (DOE, 1991)). As stated in Section J.4 of this appendix, models are simplifications of reality and, in general, several alternative models may be consistent with available data. Uncertainty in scenarios is the result of a lack of knowledge about the future of the site. Parameter uncertainty results from incomplete knowledge of the model coefficients.

The NRC's risk-informed approach to regulatory decision-making suggests that an assessment of uncertainty be included in estimating doses. Specifically, the Probabilistic Risk Assessment (PRA) Policy Statement (60 FR 42622, August 16, 1995) states, in part, "The use of PRA technology should be increased in all regulatory matters to the extent supported by the state of the art in PRA methods and data, and in a manner that complements NRC's deterministic approach and supports the NRC's traditional defense-in-depth philosophy." In the past, dose assessments in support of NRC decommissioning requirements have primarily included the use of deterministic analyses. The deterministic approach has the advantage of being simple to implement and easy to communicate to a nonspecialist audience. However, it has a significant drawback in not allowing consideration of the effects of unusual combinations of input parameters and by not providing information on uncertainty in the results, which would be helpful to the decisionmaker. Furthermore, a deterministic analysis that had a high assurance of not being exceeded would have to rely on the use of pessimistic estimates of each parameter of the model, often leading to overly conservative evaluations. Even with the use of probabilistic analyses, it is generally recognized that not all sources of uncertainty can be considered in a dose assessment nor need to be considered. The primary emphasis in an uncertainty analysis should be to identify the important assumptions and parameter values that, when altered, could change the decision.

A sensitivity analysis performed in conjunction with the uncertainty analysis can be used to identify parameters and assumptions that have the largest effect on the result. Sensitivity analysis provides a tool for understanding and explaining the influence of these key assumptions and parameter values on the variability of the estimated dose. Once identified, additional resources can be spent on refining risk-significant parameters. Sensitivity analyses also are useful in focusing NRC staff reviews on those issues most important to the decision-making process.

### I.7.2 Issues in Uncertainty/Sensitivity Analyses

Uncertainty analysis imparts more information to the decisionmaker than deterministic analysis. It characterizes a range of potential doses and the likelihood that a particular dose may be exceeded.

An important issue in uncertainty and sensitivity analysis is that not all sources of uncertainty can be easily quantified. Of the three primary sources of uncertainty in dose assessment analyses, parameter uncertainty analysis is most mature. However, approaches for quantifying

conceptual model and scenario uncertainty are less well developed. Difficulties in predicting the characteristics of future society, especially those influencing exposure, can lead to large uncertainties. At most, one is able to assert that the assessment has considered an acceptably complete suite of scenarios (Flavelle, "A Quantitative Measure of Model Validation and Its Potential Use for Regulatory Purposes" (Flavelle 1992)). For these reasons, we make no attempt to quantify formally model or scenario uncertainty, although to a certain extent, these are captured in parameter uncertainty analyses. Choices of the exposure scenarios and conceptual model(s) to be used for the site are discussed in Sections I.3 and I.4, of this appendix, respectively.

Uncertainty analyses frequently use the Monte Carlo method. Input variables for the models are selected randomly from probability distribution functions, which may be either independent or correlated to other input variable distributions. Critics of formal uncertainty analysis have often pointed out that limitations of knowledge about the nature and extent of correlation among variables fundamentally limit our ability to make meaningful statements about the degree of uncertainty in dose assessments (Smith et al., "The Effect of Neglecting Correlations When Propagating Uncertainty and Estimating the Population Distribution of Risk" (Smith et al., 1992)).

Because the results of an uncertainty analysis provide a distribution of doses, it should be recognized that some percentage of the calculated doses may exceed the regulatory limit. A key issue that should be addressed in the treatment of uncertainty is specifying how to interpret the results from an uncertainty analysis in the context of a deterministic regulatory limit. Agency practice has not been to require absolute assurance that the regulatory limit will be met, so regulatory compliance could be stated in terms of a metric of the distribution such as the mean, or a percentage of calculated doses allowed to exceed the limit. Even for a deterministic analysis, it is recognized that the reported dose is simply one of a range of possible doses that could be calculated for the site; therefore, there is still an issue of where this calculated dose should lie in terms of the unquantified spectrum of possible doses.

In summary, the key issues in addressing uncertainty are (1) incorporating alternative conceptual models and scenarios to identify a complete suite of possibilities, (2) determining how to select appropriate parameter distribution and ranges, along with the associated correlation between parameters for the analysis, and (3) specifying the metric of the dose distribution to use in determining compliance with the dose limit.

### I.7.3 Recommended Approach

#### *I.7.3.1 Screening Analyses*

Often the first step in evaluating site compliance should be a screening analysis. At preliminary stages of the evaluation, there may be little information available about the site. Therefore, the NRC's screening approach is designed to ensure that there is high confidence that the dose has not been underestimated. As discussed in Sections I.3 and I.4 of this appendix, the models and exposure scenarios used in screening were selected to represent generic conditions and are intended to be "prudently conservative." The screening analysis assumes that all that is known about a site is the source term. Accordingly, the default parameters were selected to make it unlikely for the dose that would be calculated using site-specific information to exceed the screening dose.

To develop screening values, the NRC staff performed a Monte Carlo analysis, using the DandD Version 1 code, with values of the input parameters sampled from wide ranges selected to represent the variability in those parameters across the United States. The default values of input parameters for the DandD code (i.e., the values that the code would use without specification by the user) were then chosen from distributions of those parameters based on the 90th percentile of the output dose distribution from the Monte Carlo analysis, as illustrated in Figure I.10 (NRC, 1999a).

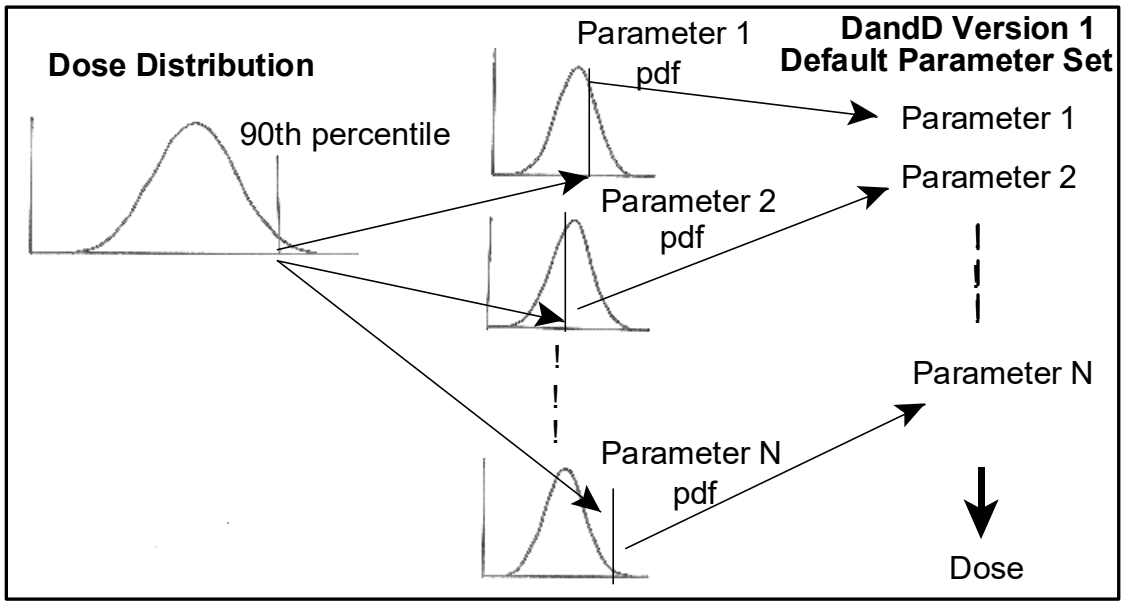


**Figure I.10 Treatment of Parameter Uncertainty in DandD Version 1**

The intent of the specification of default parameter values, exposure scenarios, and conceptual models in the DandD code was to ensure that there should be less than a 10 percent probability that the calculated dose using site-specific information would exceed the dose limit. Because the default parameters, exposure scenarios, and conceptual models in DandD Version 1.0 were designed to provide high confidence that the dose would not be underestimated, a licensee using the screening criteria did not need to quantify the uncertainty in the dose analysis. The calculated results may be considered to represent a "prudently conservative" estimate of the dose (i.e., the calculated dose is likely an overestimation of the true dose). In many cases, however, the default parameter values chosen were highly conservative, making the outcome of the deterministic analysis overly stringent.

DandD Version 2 was designed to allow Monte Carlo analysis, which gives a distribution of doses, as illustrated in Figure I.11. The code automatically performs the probabilistic analyses and aggregates the results for the user. To maintain consistency in approaches used for Versions 1 and 2, and previously published screening tables, the 90th percentile of the dose distribution was used to determine screening values that demonstrate compliance with the unrestricted release criteria in 10 CFR Part 20, Subpart E. Default parameter PDFs were incorporated into the code for screening analyses; therefore, for those, the license reviewer may only need to ensure that these aforementioned default parameters were used. See Appendix H

for additional details on the screening values and the approaches used to develop the screening values.

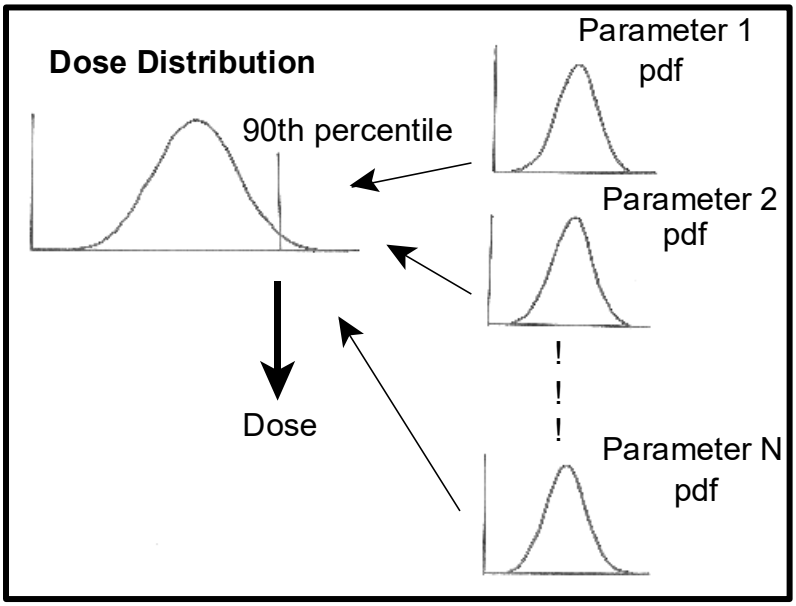


**Figure I.11 Treatment of Parameter Uncertainty in DandD Version 2**

*I.7.3.2 Site-Specific Analyses*

**DETERMINISTIC ANALYSIS**

For site-specific analyses, the treatment of uncertainty in deterministic and probabilistic analyses should be handled differently. The NRC's risk-informed approach to regulatory decision-making suggests that an assessment of uncertainty should be included in dose analyses. However, in some cases, such analyses may not be needed (e.g., bounding type analyses). Because no information is provided on the uncertainty in bounding analyses, it is important for the licensee to demonstrate that the single reported estimate of the peak dose is likely to be an overestimation of the actual peak dose. Use of conservatism in only some aspects of the analysis may not necessarily result in a conservative estimate of the dose. Uncertainties in the conceptual model may be larger than uncertainties in the parameters used in the analysis; therefore, use of conservative parameter values does not necessarily ensure a conservative estimate of the dose. To ensure that the results from a deterministic analysis are unlikely to underestimate the dose, the NRC recommends that the licensee use the approaches discussed in Sections I.3 and I.4 of this appendix for developing land use exposure scenarios and conceptual models. In addition, the licensee should use conservative values for key parameters and should use the approaches discussed below on performing sensitivity analyses when identifying key parameters in the analysis.

## PROBABILISTIC ANALYSIS

Although bounding analyses are a good starting point for determining regulatory compliance, the demonstration that a single, deterministic result is bounding may be too difficult to support. For site-specific probabilistic analysis, the licensee may use the peak of the mean dose to demonstrate regulatory criteria have been met.

A single deterministic calculation using the mean values of parameters is unlikely to result in the mean dose.

Parameter uncertainty analysis provides a quantitative method for estimating the uncertainty in calculated doses, assuming that the structure of the model is an adequate representation of the real world and that the exposure scenario is an appropriate reflection of potential future land use at the site. Several methods have been developed for quantifying parameter uncertainty, including (1) analytical methods, (2) Monte Carlo methods, (3) response surface methods, and (4) differential methods (DOE/LLW-100, "Guidelines for Sensitivity and Uncertainty Analysis of Performance Assessment Computer Codes, National Low-Level Waste Management Program," issued September 1990 (DOE, 1990)). In addition, alternative approaches, such as the first-order reliability method, have recently been applied on a wide variety of environmental problems (DOE, 1998). Of these methods, the Monte Carlo methods are recommended, because they are easy to implement and provide significant versatility.

Monte Carlo methods can be applied to either linear or nonlinear models and analytical or numerical models. Input parameter uncertainties are represented as PDFs. Parameter values randomly sampled from PDFs are used as inputs to multiple runs or "realizations" of the model.

For probabilistic analyses, the peak of the plot of mean dose over time should be compared with the regulatory standard to determine compliance. Equation 8 shows how the mean dose as a function of time can be derived. For Monte Carlo Runs:

$$Mean(t_i) = \frac{\sum_{k=1}^{N} Dose_k(t_i)}{N} \qquad \text{Equation 8}$$

where $Mean(t_i)$ = mean dose at time $t_i$
$Dose_k(t_i)$ = dose at time $t_i$ for run $k$
$t_i$ = time in years
$i$ = time steps (1 to 1000)

Essentially, a mean dose is determined at each discrete time in the analysis. A plot is then made of these means over time. The mean dose provides the "best estimate" of dose at each discrete time. The overall peak of these best estimates is then used to determine compliance with the rule. Figure I.12 shows how such a plot would be used to determine compliance with the regulations.

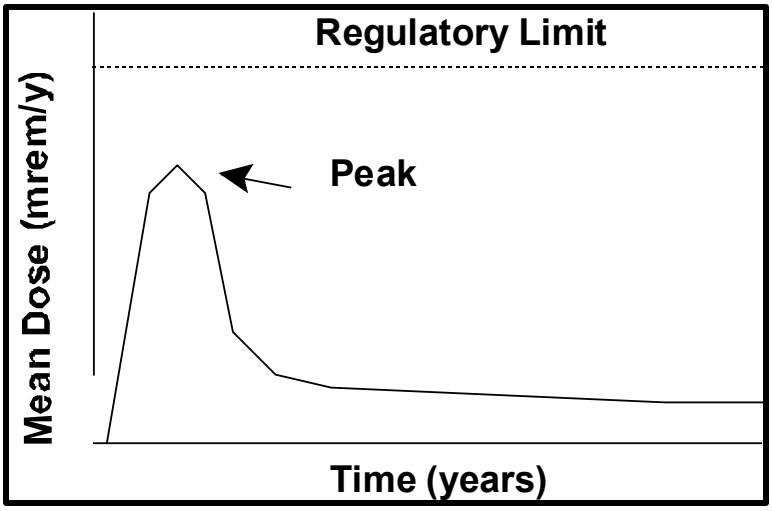


**Figure I.12 Application of "Peak of the Mean" Dose.**

Licensees using probabilistic dose modeling should use the "peak of the mean" dose for demonstrating compliance with the 10 CFR Part 20, Subpart E. Similar to all regulatory guidance, this NUREG report contains one approach for determining compliance with the regulations using probabilistic analyses. Use of "mean of the peaks" is also acceptable for demonstrating compliance. If the "mean of the peaks" dose is significantly higher than the "peak of the mean" dose, then "risk dilution" may be an issue in the probabilistic model. Consult Appendix Q for more information on the potential for and impacts of "risk dilution." If the licensee intends to use any probabilistic approach to calculate DCGLs, it should discuss its planned approach with the NRC staff.

### I.7.4 Input Parameter Distributions for Monte Carlo Analysis

A key aspect of any Monte Carlo analysis is defining the ranges and statistical distribution of parameters treated as uncertain in the analysis. It is important for the licensee to avoid assigning overly restrictive ranges that suggest an unwarranted precision in the state of knowledge. On the other hand, the specification of unreasonably large ranges may not account for what is known about a parameter and also may lead to "risk dilution." The distributions used in the analysis should characterize the degree of belief that the true but unknown value of a parameter lies within a specified range of values for that parameter.

Sensitivity results are generally less dependent on the actual distributions assigned to the input parameters than they are on the ranges chosen for the parameters. However, distributional assumptions can have a large impact on the dose distribution (SNL, SAND90-7103, "Sensitivity Analysis Techniques and Results for Performance Assessment at the Waste Isolation Pilot Plant" [SNL, 1991]). Resources can often be used most effectively by performing a Monte Carlo analysis in an iterative manner. Initially, rather crude ranges and distribution assumptions can be used to determine which input variables dominate the behavior of the calculated dose. Often, most of the variation in the calculated dose is caused by a relatively small subset of input parameters. Once the licensee identifies the most important input parameters, it can concentrate resources on characterizing their uncertainty. This avoids spending a large effort characterizing the uncertainty in parameters that have little impact on the dose (SNL, 1991).

A reasonable strategy for assigning distributions for parameters used in Monte Carlo analyses is summarized below (NUREG-1573, "A Performance Assessment Method for Low-level Waste Disposal Facilities: Recommendations of NRC's Performance Assessment Working Group," issued October 2000 (NRC, 2000b):

- **Select parameters to be assigned distributions**—Not all parameters of the system under study require specification of a distribution. Those parameters that may well be distributed, but ultimately have little impact on the results, can be assigned constant values. Even if a parameter is known to have a significant effect on the results, its value may be specified at a constant value if it can be demonstrated that the choice leads to a conservative result.

- **Assign distributions for important parameters**—The assignment of parameter distributions is usually a matter of the quantity of available data.

- **Ample data available**—Where there are ample data, empirical distributions of a parameter can be generated directly.

- **Sufficient data available**—Data plotted as histograms or in probability coordinates can be used to identify standard distributional forms (e.g., normal, lognormal, and uniform).

- **Parameters with some data**—Where there are insufficient data to estimate the shape of an empirical distribution, data may be supplemented by other soft information. For example, if there were a mechanistic basis for assigning a given distribution, or if a distribution were well known for the parameter, on a regional basis, this information could be used to estimate the likely shape of the distribution. Alternatively, the new data can be used to supplement *a priori*, non-site-specific parameter distribution (e.g., Bayesian updating).

- **Parameters with insufficient information**—If sufficient data are not available but there were other kinds of data that imply the likely behavior of a parameter, then it may be possible to supplement the desired data indirectly. An example of such a procedure is the use of root uptake factors to infer distribution coefficients in soil (Oak Ridge National Laboratory (ORNL)-5786, "A Review and Analysis of Parameters for Assessing Transport of Environmentally Released Radionuclides Through Agriculture," issued September 1984 (ORNL, 1984). Although use of root uptake factors to derive distribution coefficients for soil is provided as an example, inferred correlations between parameters should be adequately supported. If only incomplete information is known about the parameter (e.g., its mean, or its range), and no correlations to other types of data are available, then the choice of the parameter distribution should reflect the uncertainty. The distribution should have the least-biased value, which is generally a wide distribution encompassing all the possible values. One procedure to ensure that the distribution has the least bias is known as the "maximum entropy formalism," based on Shannon's informational entropy (M.E. H "Reliability Based Design in Civil Engineering" (Harr, 1987)). This formalism allows the investigator to pick the distribution based on the kinds of information available on the parameter to ensure that the result is the least biased; for example, if only the range of the data is known, a uniform distribution over the range is least biased. Table I.13 describes the maximum entropy solutions for several classes of data (Harr, 1987). Other, empirical sources of guidance for choosing parameter distributions can be found in several other references (International Atomic Energy Agency (IAEA)-SS-100, "Evaluating the Reliability of

Predictions Made Using Environmental Transfer Models" (IAEA 1989); National Council on Radiation Protection and Measurements, "A Guide for Uncertainty Analysis in Dose and Risk Assessments Related to Environmental Contamination," dated May 10, 1996 (NCRP 1996a)).

- **Parameter correlations**—Many of the parameters used in the probabilistic analyses may be correlated to other parameters. Some parameter distributions may, in fact, be used to derive other distributions (e.g., root uptake factors may be used to derive soil distribution coefficients). Also, correlations are expected on physical grounds, such as the relationship between hydraulic gradient and permeability. Where available, these correlation coefficients can then be used to generate correlated values of distributed parameters. This may help to avoid the situation where two correlated quantities are treated as uncorrelated, leading to unlikely combinations of parameters (e.g., high gradient and high-hydraulic conductivity). The effects of assumed minimum versus assumed maximum levels of correlation can be investigated to evaluate the importance of including an explicit estimate of dependency between model parameters. In some cases, explicit modeling of the dependency between model parameters is possible, based on knowledge about the explicit mechanistic reasons for the dependencies. In general, it is more important to consider the effect of dependency when correlations are strong among the model's most sensitive parameters (see discussion below on identifying sensitive parameters); weak correlations between sensitive parameters and strong correlations among insensitive parameters will generally have very little impact on the overall calculated dose (NCRP, 1996a).

**Table I. 13 Maximum Entropy Probability Distributions** *(Adapted from Harr, 1987)*

| Given Constraints on Data | Assigned Probability Density |
|---|---|
| Minimum and maximum only | Uniform |
| Expected value only | Exponential |
| Expected value and standard deviation | Normal |
| Expected value, standard deviation, minimum and maximum | Beta |
| Mean occurrence rate between arrival of independent events | Poisson |

### I.7.5 Sensitivity Analysis

Uncertainty and sensitivity analyses are closely linked, and ideally, they should be considered together. The primary aim of a sensitivity analysis is to identify the input parameters that are the major contributors to the variation or uncertainty in the calculated dose. Identifying these key parameters is essential for building a defensible case in support of the assessment. It is very important for the licensee to justify the value or range of values used in the assessment to represent these key parameters. Several of the more popular sensitivity methods used in other performance assessments conducted at the NRC are presented, very briefly, below (NRC, 1999). It may be necessary for the licensee to use more than one approach in identifying the key parameters.

The licensee should focus on the pathways and radionuclides that are providing the greatest dose. If these pathways are modified or eliminated, the sensitivity analysis should be reevaluated to verify the important parameters for the analysis, consistent with the iterative nature of the "Decommissioning and License Termination Framework" (see Section 1.5 of this volume). For sites with a suite of radionuclides, the licensee may use expected concentrations or relative ratios of radionuclides to focus resources on the overall critical pathways and parameters. In addition, the licensee should evaluate the effects of uncertainty on the relative ratios.

### *I.7.5.1 Deterministic Sensitivity Analysis*

Two types of sensitivity analysis techniques are widely used: deterministic and Monte Carlo. The first, deterministic sensitivity analysis, calculates the change in the output result (i.e., peak dose) with respect to a small change in the independent variables, one at a time. The following formula illustrates the normalized sensitivity coefficient calculated from a deterministic analysis.

$$S_i = \left[ \frac{\overline{X}_i}{d(\overline{X}_i)} \right] \left( \frac{\partial d}{\partial \overline{X}_i} \right) \qquad \text{Equation 9}$$

where $S_i$ = sensitivity coefficient

$\overline{X}_i$ = baseline value of the $i^{th}$ parameter

$d(\overline{X}_i)$ = peak dose for the baseline case

$\partial d$ = change in peak dose

$\partial X_i$ = change in $i^{th}$ parameter

Variable transformations, such as *normalization,* used in this example, are described further below.

The advantage of the deterministic technique is that it is unambiguous in terms of demonstrating a cause and effect for the given conceptual model. The disadvantages are that at least one evaluation of the model should be performed for every independent variable, and the sensitivity result applies only locally (i.e., for one location in the space of all of the independent variables).

### *I.7.5.2 Statistical Sensitivity Analysis Techniques*

The techniques used herein (except deterministic analysis) rely on the use of the Monte Carlo method for probabilistically determining system performance. Statistical analysis of Monte Carlo results starts with a large pool of realizations (hundreds to thousands). These techniques determine sensitivities of the dependent variable (dose) to changes in the independent variables. The main advantage of these techniques is that they allow sensitivity to be determined over wide ranges of the independent variables, as opposed to the deterministic techniques that apply to only one point within the ranges. The disadvantage of statistical techniques is that it is often difficult to extract useful information on sensitivity except for a small set of the most important variables, because smaller sensitivities are obscured. A compilation of some of the more popular techniques for analyzing sensitivity from Monte Carlo results is presented below.

Usually, statistical sensitivity techniques have been applied to the set of peak doses drawn from the realizations. Sensitivity information from the ensemble of the peak doses is useful and is consistent with the "mean of the peaks" dose. For the "peak of the mean" dose, the set of doses drawn from the Monte Carlo runs at the time of the "peak of the mean" dose could be used.

## SCATTER PLOT AND LINEAR REGRESSION ON ONE VARIABLE

In the scatter plot/single linear regression technique, peak TEDE is plotted versus each of the sampled input variables. This is often a good starting point for examining Monte Carlo results, because strong relationships between peak dose and the independent variables are often obvious. Single linear regression of Monte Carlo results may fail to show an unambiguous correlation, since other sampled parameters that affect the output are varying at the same time.

## USE OF THE T-STATISTIC TO DETERMINE SIGNIFICANCE OF SINGLE LINEAR REGRESSION PARAMETERS

The t-test estimates the confidence that an estimated parameter value differs from another value. In this case, it is used to determine if there is a specified (e.g., 95-percent) confidence that the slope ($m_i$) of a single linear regression is different from zero (Benjamin and Cornell, "Probability, Statistics, and Decision for Civil Engineers" (Benjamin and Cornell, 1970).

The t statistic of the slope of the regression line is defined:

$$t_i = m_i \sqrt{n \frac{S_i^2, x}{S^2}} \qquad \text{Equation 10}$$

where $t_i$ = t-statistic for regression coefficient $i$
$m_i$ = estimated value of regression coefficient (i.e., slope of the best-fit line for dose versus the independent variable $i$)
$S$ = estimated standard deviation of dose
$S_{i,x}$ = estimated standard deviation of independent variable $x_i$
$n$ = number of samples

When the number of realizations is large, the t distribution may be represented by the normal distribution. The critical value to ensure 95-percent confidence that $m_i$ differs from zero under these conditions is 1.96. Equation 10 is used, therefore, to determine whether the absolute value of the $t$ statistic for each independent variable is greater than 1.96. If not, then the hypothesis that the independent variable is significant is rejected.

## PARTIAL RANK CORRELATION

The partial rank correlation coefficient measures the strength of the relationship between variables after any confounding influences of other variables have been removed. The partial rank correlation coefficient between $X_1$ and $Y$, with the influence of $X_2$ removed, is given by:

$$\rho(X_1 Y X_2) = \frac{\rho_{X_1 Y} - \left(\rho_{X_1 X_2}\right)\left(\rho_{Y X_2}\right)}{\left[\left(1 - \rho_{X_1 X_2}{}^2\right)\left(1 - \rho_{Y X_2}{}^2\right)\right]^{1/2}} \qquad \text{Equation 11}$$

where $\rho(X_1 Y X_2)$ = partial rank correlation coefficient between $X_1$ and $Y$, with the influence of $X_2$ removed
$\rho_{X1Y}$ = rank correlation coefficient between $X_1$ and $Y$
$\rho_{X1X2}$ = rank correlation coefficient between $X_1$ and $X_2$
$\rho_{YX2}$ = rank correlation coefficient between $Y$ and $X_2$

### Stepwise Multiple Linear Regression

Stepwise multiple linear regression (stepwise regression) determines the most influential independent variables on output uncertainty according to how much each reduces the residual sum of squares (RSS) (SNL, 1991). The form of the regression equation is:

$$y = m_1 x_1 + m_1 x_1 + \ldots + \quad m_n x_n + b \qquad \text{Equation 12}$$

where $y$ = dependent variable (i.e., peak dose)
$x_i$ = independent variables
$m_i$ = regression coefficients
$b$ = intercept

The variables may be the raw variables, transformed variables (e.g., logarithms), or ranks (see Section I.7.5.3.2 of this appendix). The stepwise algorithm calculates the reduction in RSS for the independent variables in the order that gives the greatest reduction first. The regression coefficients ($m_i$) are the partial derivatives of the dependent variable with respect to each of the independent variables; therefore, $m_i$ provides a measure of the relative change in output with respect to a change in the input variable, given that the other input variables are held constant.

## NONPARAMETRIC TESTS

Nonparametric tests differ from regression and differential analyses in that they do not require fitting the data to prespecified functional form. The Kolmogorov-Smirnov test is one such test that determines whether a set of samples has been drawn from a specific distribution (NRC, 1988a). It is used to determine whether an independent variable is important by comparing a subset of the independent variable composed of the values from the highest category (e.g., 10 percent) of the peak TEDE realizations to the theoretical distribution of that independent variable. If the distributions are equivalent, then peak TEDE is not sensitive to the variable in question. Conversely, if the distributions are different, then the variable in question does have an effect on peak TEDE.

### *I.7.5.3 Variable Transformations and Their Attributes*

Demonstrating the relationship among input and output variables can be enhanced by transforming the variables. This section describes some common variable transformations used in sensitivity analysis.

**NORMALIZATION**

In normalization, the input variable $x_i$ is transformed by dividing by its mean value (or another baseline, such as the median or the 90th percentile):

$$X_i^* = \frac{X_i}{\overline{X_i}} \qquad \text{Equation 13}$$

Normalized variables are dimensionless and are scalar multiples of their baseline values. Dimensionless variables allow the comparison of sensitivities to other independent variables with different dimensions. Normalized variables are a natural outcome of sensitivity derived from regression of log-transformed variables. Such sensitivity measures describe only the relative change in the dependent variable (peak TEDE) to changes in the independent variables. Sensitivities calculated from normalized variables do not take into account the uncertainty in the independent variables.

**RANK TRANSFORMATION**

Rank transformation, a dimensionless transform, replaces the value of a variable by its rank (i.e., the position in a list that has been sorted from largest to smallest values) (Iman and Conover, "The Use of Rank Transform in Regression" (Iman and Conover 1979)). Analyses with ranks tend to show a greater sensitivity than results with untransformed variables and diminish the influence of the tails in highly skewed distributions.

**LOGARITHMIC TRANSFORMATION**

For situations in which input and output variables range over many orders of magnitude, it may be advantageous or even necessary to perform analyses on the logarithm of the variables instead of the variable values themselves. The log transformation is also valuable for creating regression equations, where the subprocesses of the model multiply each other to form the output variable. For the present situation, in which the dose calculation results from radionuclide releases from the waste form, transport through the geosphere, and uptake by humans, the processes are indeed largely multiplicative rather than additive. Log transforms, therefore, tend to give better fits to the Monte Carlo results than untransformed variables. The log transformation is generally used in conjunction with normalization.

**STANDARDIZATION**

The independent and dependent variables can be standardized by subtracting the mean and dividing by the standard deviation; that is,

$$x_i^* = \frac{x_i - \bar{x}}{\sigma_x}$$ Equation 14

The advantage of standardization over normalization is that it inserts the approximate range of the variables into the sensitivity analyses. Therefore, a variable that has a large per-unit sensitivity, but is well known and has a narrow range, will have an increased sensitivity when standardized. Conversely, independent variables with wide ranges may show a reduced sensitivity when standardized.

Sensitivity measures based on standardized variables (standardized sensitivities) have the advantage of taking into account the uncertainty (in terms of the standard deviation) of the independent variable. This technique decreases the sensitivity if the range of the independent variable is large. Furthermore, the standardized sensitivities preserve the absolute values of peak TEDE, since the derivatives are divided by the standard deviation for the entire set of calculations rather than the mean peak TEDE at the evaluation point.

### I.7.6 Conclusions

Sensitivity analyses identify parameters of the models and assumptions that have the largest effect on the results. The licensee should use sensitivity results to determine if more information on key parameters is warranted to make a convincing case for the acceptability of the site for release. The sensitivity techniques discussed here evaluate sensitivity in different ways, and all have their strengths and weaknesses. A useful way to use sensitivity results is to employ several different techniques and then determine if a common set of parameters regularly turns out to be important.

Appendix Q contains additional information related to the NRC staff's observations and recommendations related to consideration of uncertainty in dose and performance assessment analyses. Included are specific issues related to data representativeness (temporal and spatial), data correlations, use of generic data, consideration of alternative conceptual models, model integration, model abstraction, and preparation of sensitivity and uncertainty analyses.

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# APPENDIX J

# ASSESSMENT STRATEGY FOR BURIED MATERIAL

## J.1 Generic Description of Situations

This appendix describes examples of situations warranting alternative conceptual models which may occur at sites undergoing decommissioning. The first situation discusses buried radioactive material or subsurface contamination at a site. The second situation discusses subsurface structures or basements which are backfilled and left as part of the end state of the site.

### J.1.1 Buried Radioactive Material or Subsurface Soil Contamination

A licensed site has the following characteristics for purposes of the conceptual model for buried radioactivity:

- It has buried radioactive material or it has subsurface soil contaminated with residual radioactivity beneath clean cover soils or fill material[1]. (Sites prior to the 1980's may have 10 CFR 20.304 burial units that were in use before the January 28, 1981, effective date of the final rule, "Standards for Protection Against Radiation; Burial of Small Quantities of Radionuclides" (45 FR 71761), when Section 20.304 was removed from the regulations).

- The site has no other sources of residual radioactivity, or the total dose associated with all sources of residual radioactivity must be addressed and be below the dose limit (see Section 2.7 of this NUREG report).

- Information on the inventory (e.g., radionuclide concentrations, disposal dates, waste form) may be limited. However, the licensee has enough information to estimate or to bound the total activity or concentrations of radioactive material present (see Section 4.0 for information on the HSA, Section 4.2 for information on characterization surveys, and Section I.2 of Appendix I for information about source term abstraction).

- It is known that the material is deep enough that an external dose at the surface is not possible in the current configuration.

- The site does not have any of the physical limitations that would make the use of a RESRAD code to represent the conceptual model inappropriate (see Appendix I, Section 4, Table I-9).

- The site is underlain by an unsaturated layer and aquifer. If the aquifer is potable, and has sufficient yield, the groundwater pathway should be considered for drinking water and/or irrigation. If the aquifer is not potable, or has insufficient yield, then the licensee may present arguments for why the groundwater (drinking water) pathway does not need to be considered, although the licensee may need to consider other uses of groundwater (e.g., irrigation).

[1] Licensees should provide support that the cover or fill materials contain no radioactive contamination. If potentially contaminated fill materials are used (e.g., reuse of rubblized concrete from the site as fill), the licensee must adequately survey the materials before reusing them for debris and consider the dose contributions of residual radioactivity present in fill materials. See Section G.3.2. for additional details on radiological surveys associated with fill materials.

- If the soil at the site is assumed to be capable of growing crops without significant soil engineering, then plant ingestion should be considered.
- Arguments may be presented for why certain pathways do not need to be considered.

Figure J.1 shows a simple conceptual figure of the site with buried radioactive material.

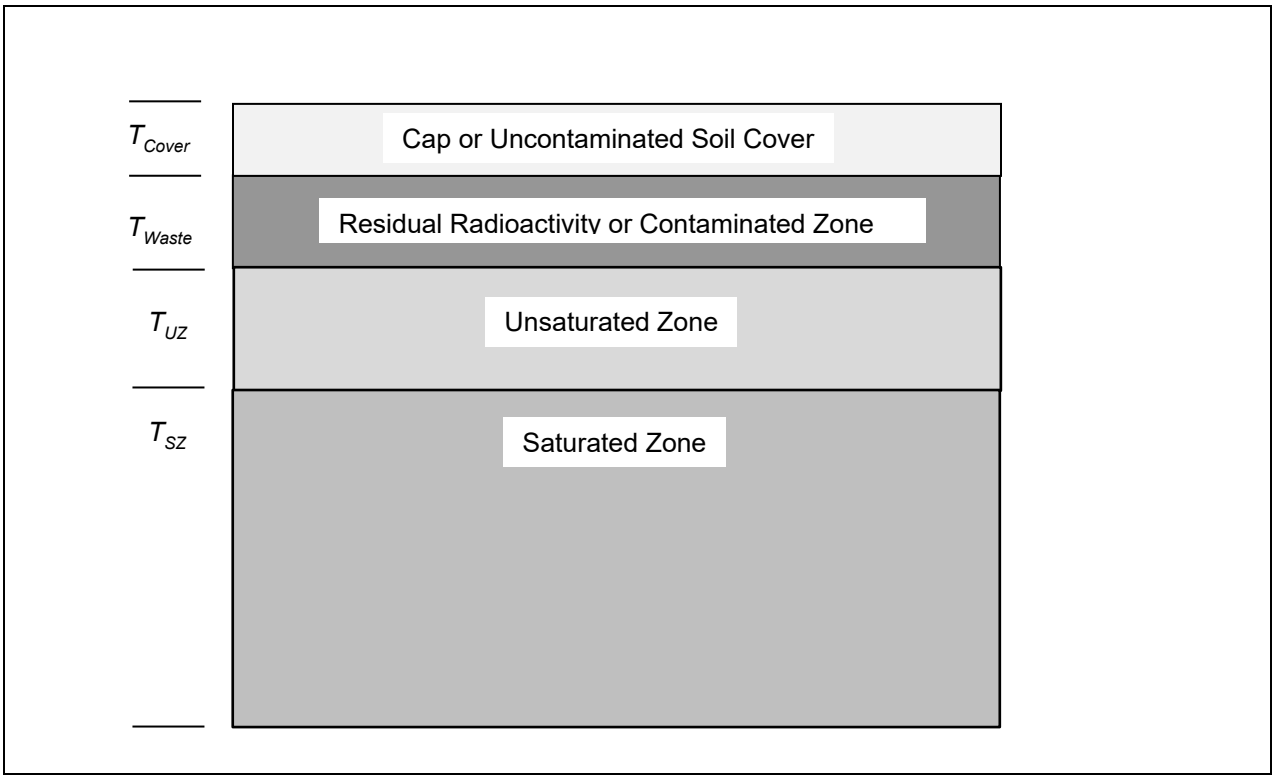


**Figure J.1 Conceptual Illustration of Buried Material in Subsurface Soil at a Site**

### J.1.2 Backfilled Basements

In some cases, the site may have below-grade basements that will be backfilled and remain after decommissioning. For these sites, there may be some residual radioactivity left on the concrete surfaces prior to backfill, and a different conceptual model is warranted to derive site-specific DCGLs for those concrete surfaces (see Figure J.2).

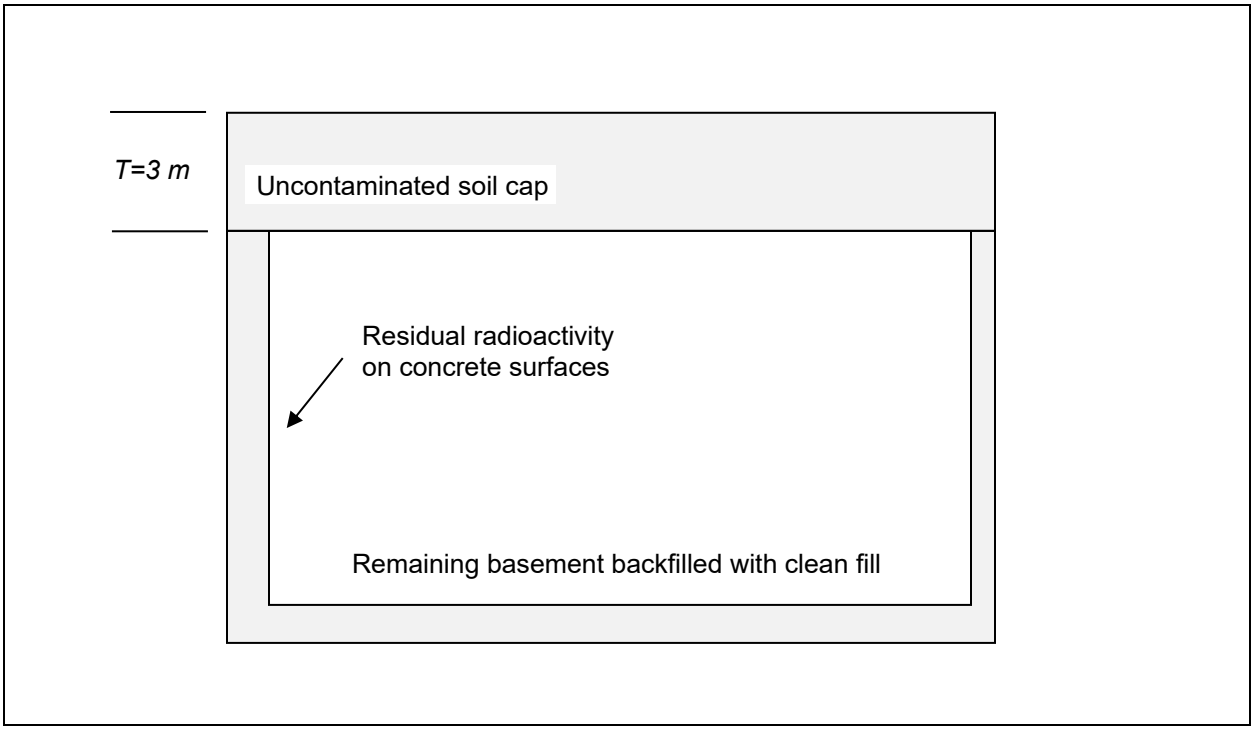


**Figure J.2 Conceptual Model for Remaining Basements which have Residual Radioactivity and are Backfilled with Clean Fill**

## J.2 Conceptual Models, Exposure Scenarios, Exposure Pathways, and Critical Group

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

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

In the absence of site-specific information, residential scenarios are generally assumed in the following examples for illustrative purposes, although arguments can be presented for why other scenarios are reasonably foreseeable and residential scenarios are less likely but plausible (or implausible). For more information on developing exposure scenarios, see Appendix I and Appendix M of this volume.

### J.2.1 Buried Material Conceptual Models, Exposure Scenario, Exposure Pathways, and Critical Group

For sites with buried material, one conceptual model could assume that all the buried radioactive material at depth was instead at the surface (Figure J.3). That is, this scenario takes no credit for the clean cover (the clean cover is removed in the conceptual model) and models leaching of the radionuclides into the groundwater which is then used by a resident, as well as considering direct radiation from exposure to the soil at the surface. Care should be taken to ensure the modeled vadose (unsaturated) zone thickness is equal to or less than the actual vadose zone thickness (i.e., the contamination zone should not be moved to the top of the cover but should remain at the actual elevation).

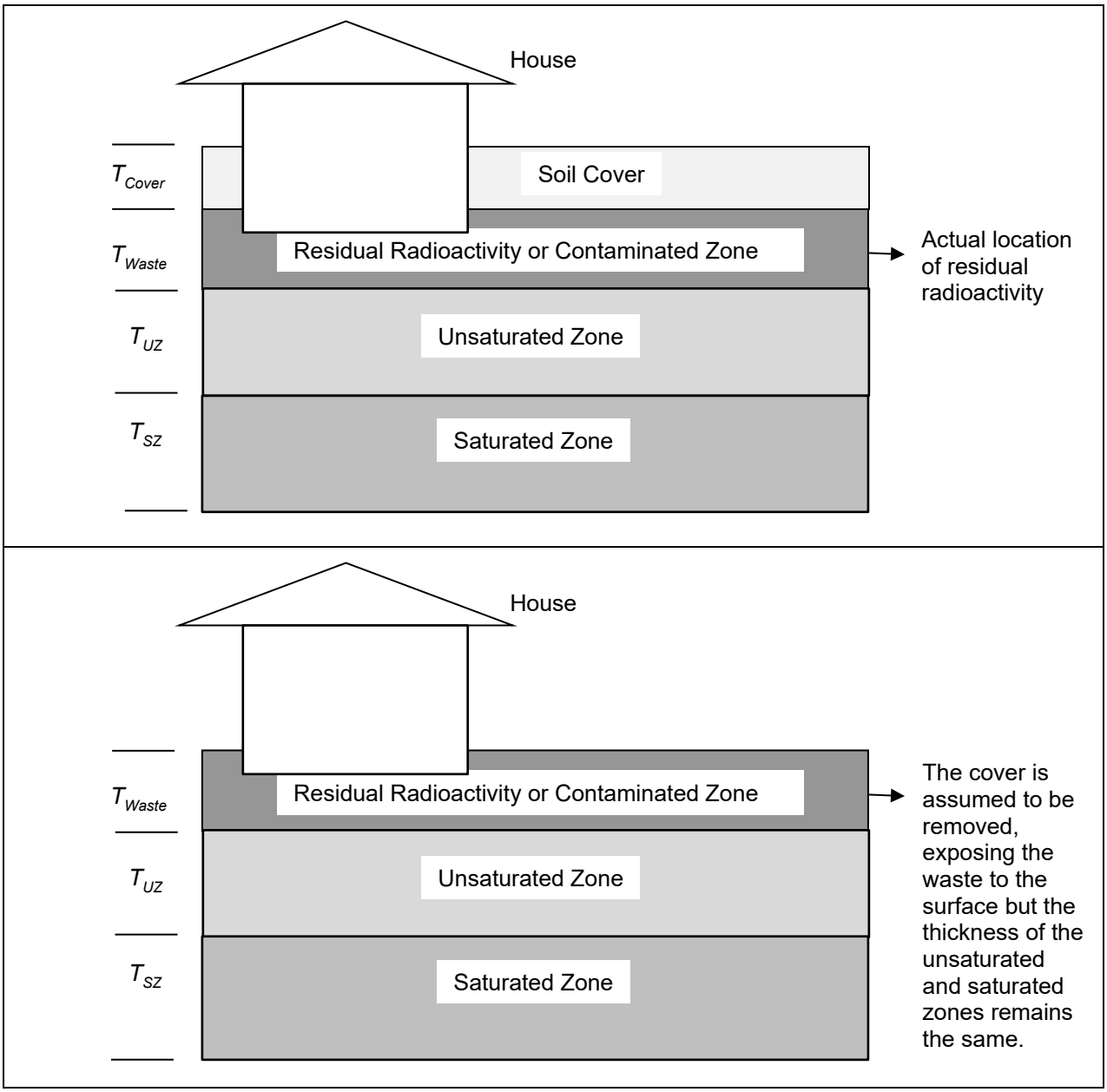


**Figure J.3 Simplified Conceptual Model of Human Disturbance into Buried Residual Radioactivity** *(the top Panel Shows the Original Configuration of Residual Radioactivity and Human Disturbance Event [Construction of a Home with Basement]; the bottom Panel Shows a Conceptual Model with the Cover Assumed to have been Removed for Simplification)*

Alternatively, credit for the cover may be taken and two exposure scenarios can be developed (1) leaching of the radionuclides from their buried position to groundwater, and (2) intrusion into the buried residual radioactivity by basement construction (or other large-scale excavation scenario) or well drilling. In the second scenario the displaced soil, which includes part of the residual radioactivity, is spread across the surface (Figure J.4 and Figure J.5). All the exposure pathways are considered for the second scenario involving intrusion into the buried residual radioactivity and, although not all the source term is in the original position, leaching will occur both from the remaining buried residual radioactivity (if there is any) and the surface soil.

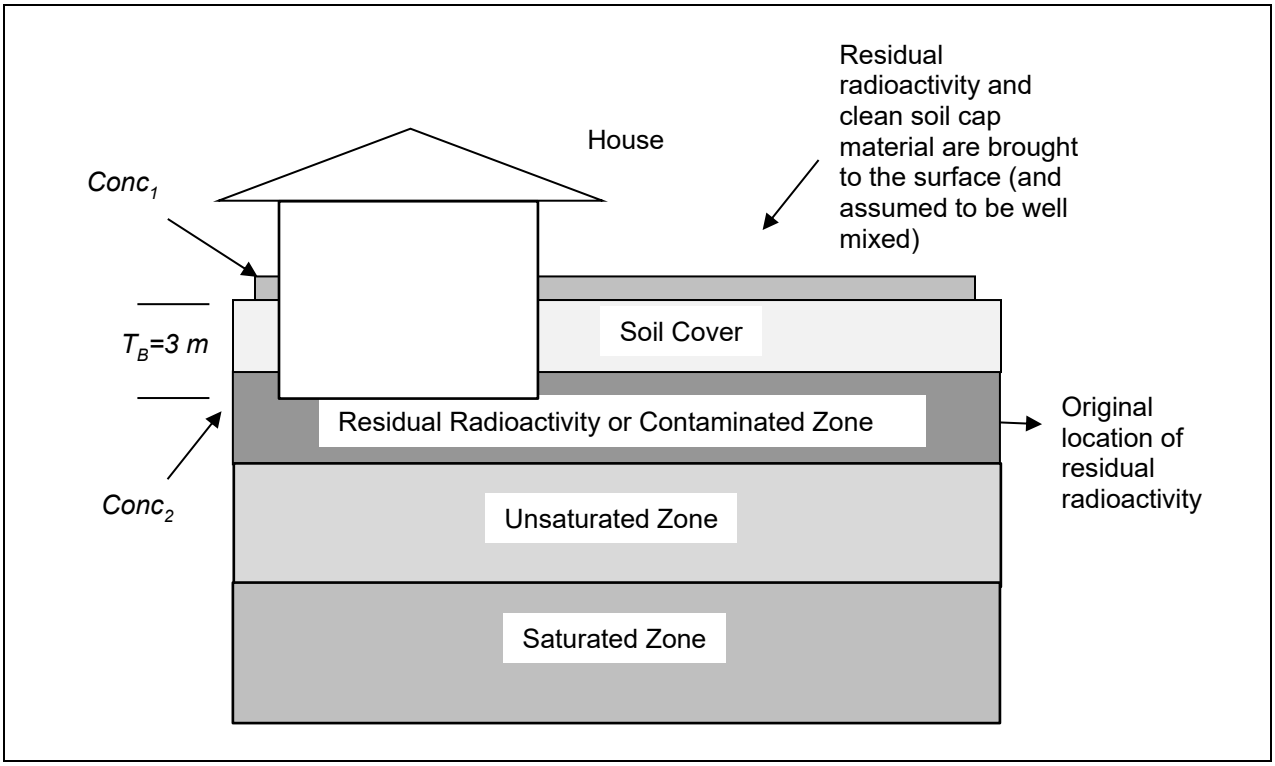


**Figure J.4 Conceptual Model of Human Disturbance into Buried Residual Radioactivity** *(Showing some of the Waste Mixed with Clean Soil on the Surface [$Conc_1$] and some Waste Remaining at Depth [$Conc_2$])*

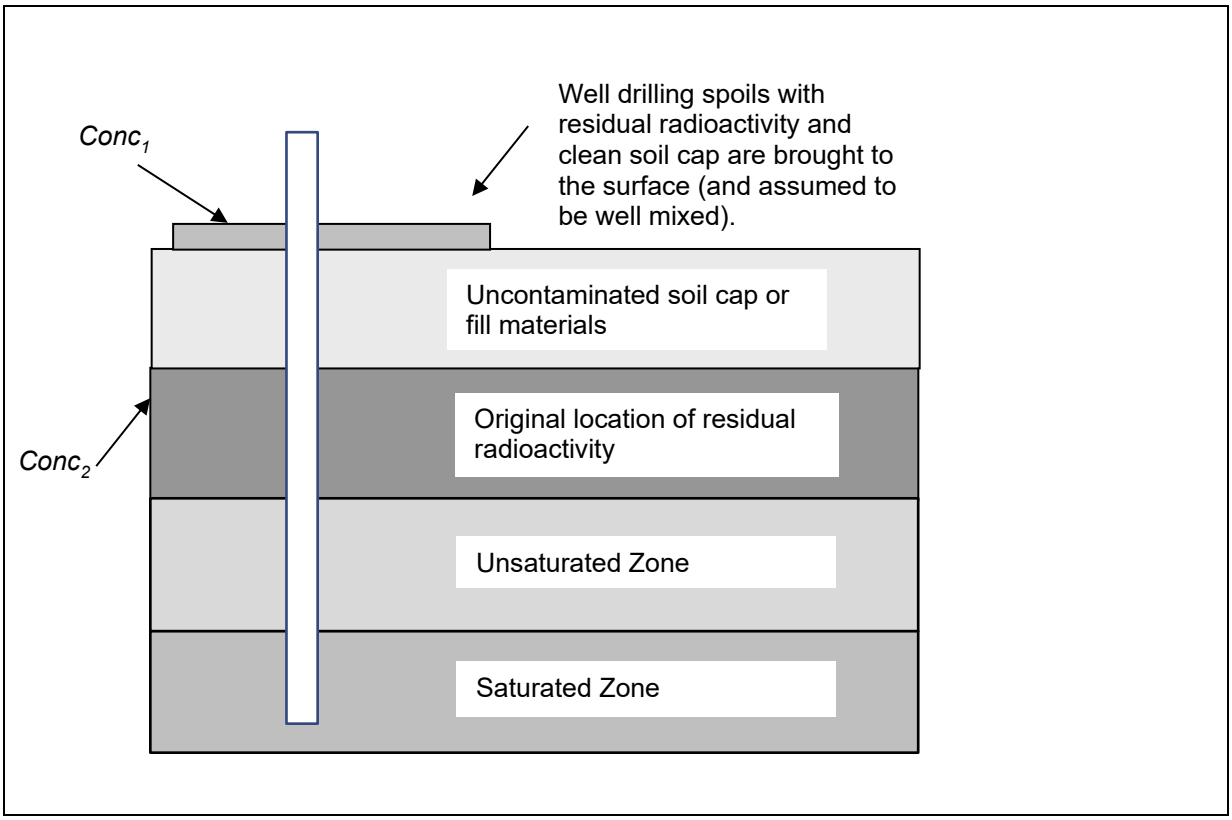


**Figure J.5 Conceptual Model of Buried Radioactivity with a Well Drilled through the Radioactive Material and the Drilling Spoils Spread out on the Surface**

### J.2.2 Additional Considerations for Backfilled Basement Conceptual Models, Exposure Scenarios and Pathways

For backfilled basements a similar approach can be considered as that for buried material. An intrusion event (well-driller scenario) that includes leaching into the groundwater can be developed. In the well-driller scenario, the backfilled basement is assumed to remain undisturbed except for a well drilled through a portion[2] of the backfilled substructure for drinking water.

The well-drilling scenario considers (i) pathways associated with use of contaminated groundwater from an onsite well and (ii) direct (and indirect) exposure to drilling spoils that are brought to the surface during the installation of the onsite well. These two exposure scenarios have different conceptual models. For the groundwater scenario (Figure J.6), licensees may be able to make simplifying assumptions if the simplifications do not underestimate the potential

[2] Because it is more pessimistic to assume that the driller stops drilling after encountering reinforced concrete (would lead to higher concentrations and dose with less clean material located below the concrete being brought to the surface and mixed with the residual radioactivity in the reinforced concrete), this assumption is acceptable. Arguments can be presented on why the driller might be able to drill through the reinforced concrete floor and bring up additional relatively clean material between the concrete floor and the drinking water aquifer. This would potentially allow cleaner soil below the concrete basement to be brought to the surface and mixed with the residual radioactivity associated with the concrete surface.

dose (e.g., licensees may be able to assume all the residual radioactivity exists as a contaminated zone layer of soil at the elevation of the floor of the underground structure closer to the water table aquifer). To simplify the conceptual model further, the concrete floor and walls can be assumed to not affect flow, allowing more rapid transport of residual radioactivity to groundwater[3]. If the simplifications lead to excessive pessimism, licensees have the option to model the actual configuration of residual radioactivity, degradation of cementitious materials, and flow and transport of residual radioactivity to groundwater. However, given the complexity and uncertainty in more realistically modeling these processes, it is expected that uncertainty will likely need to be managed with conservative assumptions. The licensee should communicate with the NRC early in the process to ensure that a technically acceptable approach is developed for these more complex problems.

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

For the well-driller direct exposure pathway (Figure J.7), a well is assumed to be installed through the fill material, and drilling spoils along with a portion of the concrete floor are brought to the surface. The well installation may occur before any leaching from concrete occurs. The residual radioactivity in the concrete of the basement floor that is contacted by the borehole during installation of the well is assumed to be mixed with the column of drilling spoil fill material above the floor surface, brought to the ground surface, and spread.

[3] If residual radioactivity associated with the basement is below the water table, the licensee should ensure that the conceptual model for the site is compatible with the conceptual model of the selected code. Commonly used decommissioning dose modeling codes assume the residual radioactivity is in the vadose zone. More complex site-specific modeling may be warranted, or adjustments can be made to existing models/codes, if the limitations of the selected codes are well understood and the adjustments can be shown to not result in an underestimation of dose. See Appendix I for additional information.

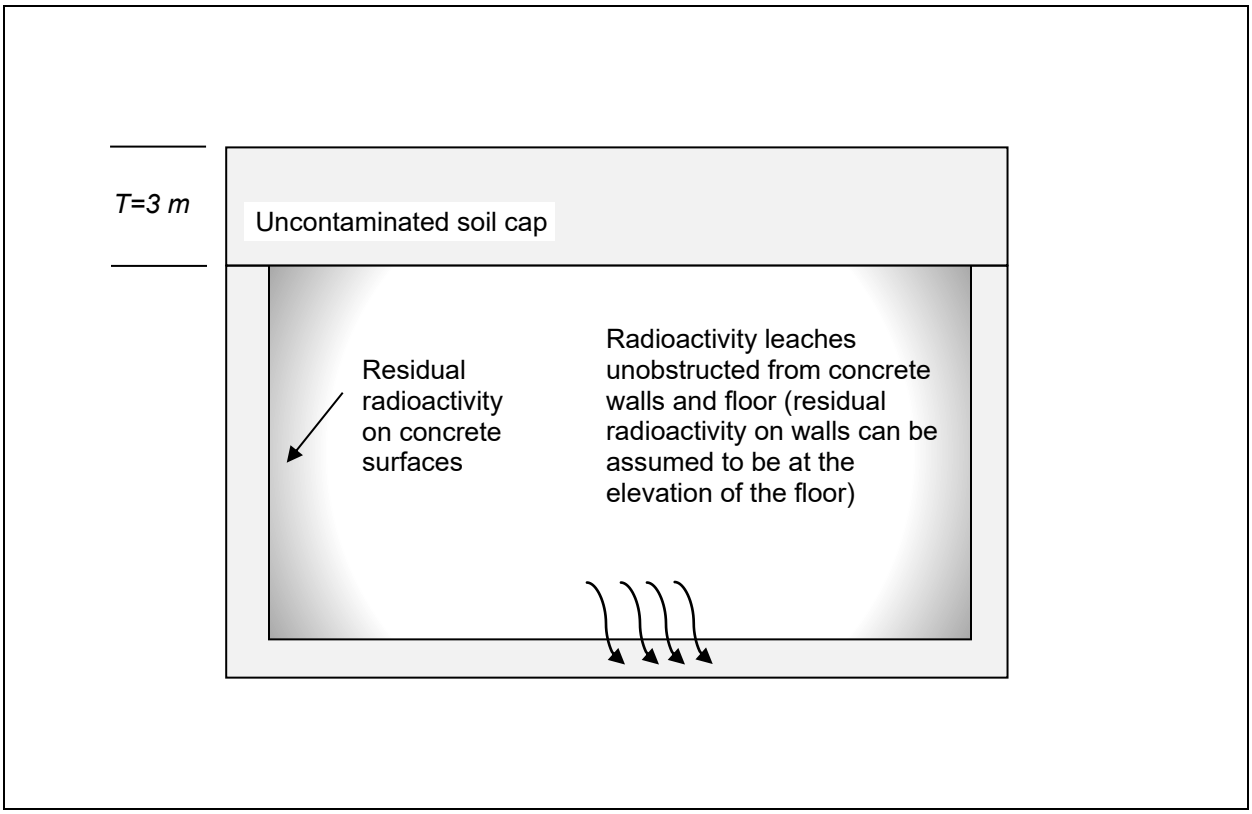


**Figure J.6 Conceptual Model for Groundwater Exposure Pathway from Remaining Basements which have Residual Radioactivity and are Backfilled with Clean Fill** *(where Residual Radioactivity is Assumed to be Released from the Concrete and Leach into the Groundwater)*

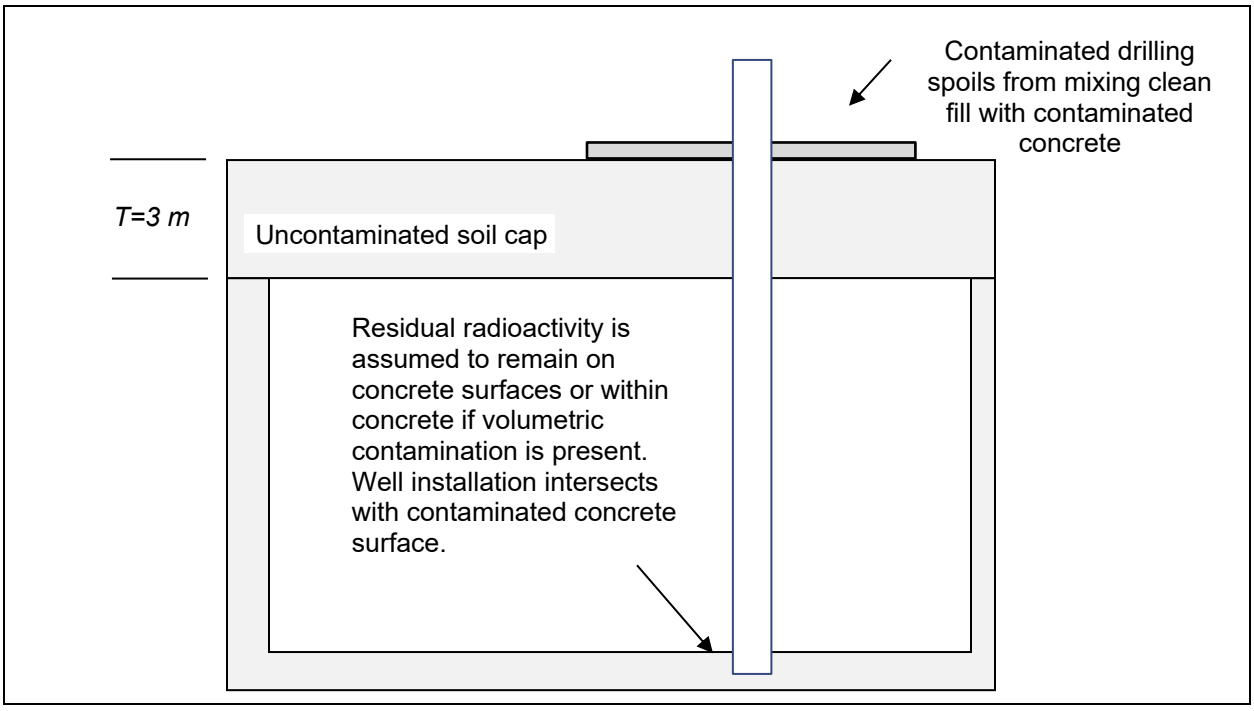


**Figure J.7 Conceptual Model for Well-Driller Scenario**

Additional scenarios that may need to be considered for large backfilled subgrade structures (e.g., containment basements, auxiliary building basements, and/or turbine basements at a reactor site) may be described as large-scale excavation[4]. A large-scale excavation scenario, which would bring up more residual radioactivity compared to a well-driller scenario, may need to be considered given the depth, geometry, and materials associated with the source. In this scenario, a portion or all the remaining backfilled basement walls and/or floor are removed in the future after license termination. In this scenario, all the residual radioactivity is assumed to remain on the concrete to maximize the concentrations in the concrete brought to the surface. An example conceptual model is shown in Figure J.8. Note that this figure is for illustrative purposes and does not represent a specific suggested geometry or scale of an excavation. The dose to the construction worker, disposal facility worker, or other member of the public who may be exposed to the residual radioactivity on the excavated concrete and/or fill material would need to be considered. The radionuclide concentrations (Bq/kg or pCi/g) in the excavated concrete is directly related to the ratio of concrete surface area to concrete volume excavated. For substructures that have thick walls and for where there the contamination is limited to the surface, assuming a partial excavation that includes only the walls with the minimum thickness would result in a higher concentration. Arguments could be presented by the licensee why a large excavation scenario is not reasonably foreseeable or why certain pathways are less likely but plausible, or implausible and are simply used to inform decision-making, or do not need to be considered (see Chapter 5 for additional information on compliance, informative, or eliminated exposure scenarios).

4 For the case of large backfilled subgrade structures, the depth of the basement is typically greater than 3 m (10 ft), and the home basement excavation scenario can be eliminated from consideration for multiple reasons (depth of waste and potential difficulty constructing a residential basement through reinforced concrete structures).

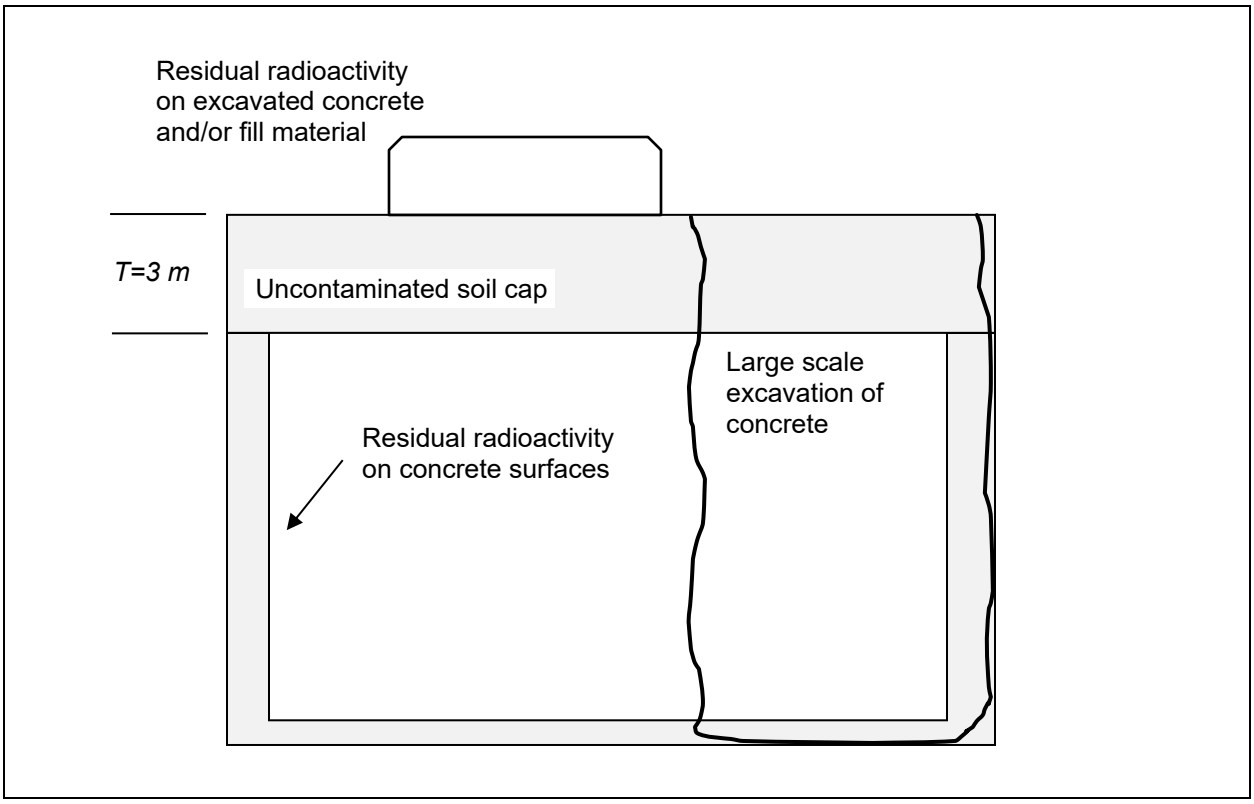


**Figure J.8 Large-Scale Excavation Conceptual Model for Remaining Basement which has Residual Radioactivity and is Backfilled**

## J.3 Examples of Analyses Using the RESRAD Family of Codes

The RESRAD computer code was developed by Argonne National Laboratory (ANL) for the U.S. Department of Energy (DOE) to calculate site-specific residual radiation guidelines and radiation dose to future hypothetical onsite individuals at sites contaminated with residual radioactive material. DOE adopted the RESRAD-ONSITE code in Order 5400.5 for derivation of soil cleanup criteria and dose calculations, and it is widely used by DOE, other Federal agencies, and industry.[5]

The RESRAD family of codes have an assumed conceptual model; therefore, the analyst only must determine if the assumed conceptual model is appropriate for the problem. However, unlike DandD, the RESRAD family of codes do not have prescribed exposure scenarios. The analyst should develop the exposure scenario by switching on or off various exposure pathways or adjusting parameters. For the standard resident farmer exposure scenario commonly used by the NRC staff, all the exposure pathways should be switched on for unrestricted use cases, with the exception of the radon pathway. The analyst should justify excluding any of the other pathways. For example, if it can be shown that the groundwater at the site cannot be used because of either poor ambient water quality (e.g., salinity) or low yields, the licensee may justify elimination of the groundwater pathway. A finding that the groundwater is unsuitable is

[5] RESRAD-ONSITE is referred to as RESRAD in DOE Order 5400.5

typically made in coordination with State agencies. For more information on eliminating pathways, see Appendix I, and Appendix M.

The RESRAD family of codes require that the radioactive inventory be input as a source concentration. Because the codes are designed for conducting site-specific analyses, it is expected that, for most analyses, the analyst will have data on radionuclide concentrations at the site.[6] It should be appropriate to use the arithmetic average of the radionuclide concentration in the analysis (note this also includes any interspersing clean soil). The RESRAD-ONSITE and RESRAD-OFFSITE codes allow the user to input information on the area and thickness of the residual radioactivity (i.e., these are not fixed, although defaults are provided). For surface residual radioactivity (≤0.9 m (3 ft), which is the default root depth), the site-specific mean concentration, area of residual radioactivity, and thickness of the residual radioactivity can be input directly in the code. For deeper residual radioactivity, or if the residual radioactivity is capped (such as with burials), assumptions should be made about how much waste may be brought to the surface and how it may be mixed with uncontaminated soil.

### J.3.1 Modeling Buried Material Using RESRAD

Analyzing the conceptual model in Figure J.4 using RESRAD requires two simulations. During the first simulation, it is assumed that a small volume of waste due to construction of a house basement (600 $m^3$ (21,200 $ft^3$)) is brought to the surface and spread out over an area to a depth of 0.9 m (3 ft). The source concentration for this first simulation could be derived by multiplying the average concentration of the buried residual radioactivity by the fraction of the total 3 m home (basement) construction excavation thickness that is assumed to contain residual radioactivity (i.e., credit is taken for mixing and dilution of the residual radioactivity in a larger volume of excavated soil, a portion of which is assumed to be free of residual radioactivity). The concentrations could take credit to account for radioactive decay. An area that could be used in the first simulation is 700 $m^2$ (7,550 $ft^2$) (i.e., 600 $m^3$ (21,200 $ft^3$) divided by 0.9 m (3 ft)). The assumed contaminated thickness would then be 0.9 m (3 ft). However, the licensee should perform a sensitivity analysis to understand the impact of assumptions on the distribution of residual radioactivity on the surface (area and thickness of the contaminated zone) to dose. If the dose results are significantly different, the licensee should consider changing the assumptions on the distribution of contamination to ensure that the doses are not significantly underestimated. The second simulation looks at effects from exposure from the remaining residual radioactivity. If the residual radioactivity is presently capped, it can be assumed to be covered for the second simulation, unless there are reasons to model the situation with the cover removed (e.g., high soil erosion rate). The source concentration for the second simulation should be the mean concentration for the waste area, including interspersing clean soil. The area and thickness of the residual radioactivity used in the second simulation would be based upon the true existing source configuration. Accordingly, to use this approach, the analyst will have to know something about the waste zone configuration. Additionally, the analyst may need to consider the geometry of the buried residual radioactivity in relation to the receptor located in the basement and other floors of the home. The RESRAD-ONSITE and RESRAD-BUILD codes generically assume that residual radioactivity is located directly underneath the residence (residual radioactivity surrounding a basement of a residence is not explicitly

[6] RESRAD-ONSITE and RESRAD-OFFSITE are primarily designed to look at radioactively contaminated soils; therefore, for analyses involving other types of wastes, the analyst will have to make some assumptions about the waste form and how the radionuclides will be released from this waste form. These assumptions should be clearly laid out.

considered). It may be possible to use a shielding code to calculate an effective shielding factor to implicitly consider such an exposure scenario.

Analyzing the conceptual model in Figure J.5 would also require a dual simulation in RESRAD like what is described for the conceptual model in Figure J.4, except the concentration and volume of the material brought to the surface is based on drilling spoils as opposed to house construction. For example, one could assume a well that has a diameter typical for residual drinking water use in the area and is sufficiently deep to reach the aquifer is drilled through the buried radioactive material. The drilling spoils are then spread across the surface to a certain thickness (e.g., 0.15 m). The licensee may take credit for mixing the buried radioactive material with the clean material when it is brought to the surface in determining the concentration of the drilling spoils. Again, the licensee should perform a sensitivity analysis to understand the impact of assumptions on the distribution of residual radioactivity on the surface (area and thickness) to dose. The second simulation analyzes the effects from leaching of the remaining residual radioactivity into the groundwater.

An alternative to using the dual simulation approach is to simply assume that the waste is uniformly distributed over the source volume, taking no credit for the cover as shown in the lower portion of Figure J.3 (i.e., by assuming that the cap is not present). This should provide a conservative estimate of the dose. The thickness of the vadose zone should reflect the actual distance to the aquifer to ensure that the travel time to the aquifer is not significantly overestimated, particularly if short-lived radionuclides are present. Using this simpler approach, the analyst would use the mean concentration as the source concentration. In using either of these approaches, it may be important to assess the appropriateness of assuming that the activity is uniformly distributed over the waste volume (e.g., if elevated areas are present, it may be appropriate to evaluate the dose contributions of the elevated areas [see Sections A.5.2 and I.2.3 for additional information on evaluation of elevated areas]).

If all that is known is the source inventory (activities), such as at some old burial sites, the source concentration of the material brought to the surface can be calculated by dividing the total inventory by the total mass of the waste (i.e., volume of the waste multiplied by density of the waste) and then multiplying this average concentration of buried residual radioactivity by the fraction of the total 3 m home (basement) construction excavation thickness that is assumed to contain residual radioactivity (i.e., credit is taken for mixing and dilution of the residual radioactivity in a larger volume of excavated soil a portion of which is assumed to be free of residual radioactivity). The concentrations could take credit to account for radioactive decay. The density used to calculate the concentration should be the same density assumed in the analysis.

### J.3.2 Modeling Backfilled Basements Using RESRAD

Analyzing the conceptual models for backfilled basements also require dual simulations in RESRAD. For example, in Figure J.7, the drilling spoils can be modeled as a contaminated zone at the surface. The well drilling may be assumed to stop once it hits the concrete floor, but the depth should be sufficient enough to bring up all surface or volumetric contamination in the concrete floor that it intersects. The concentration of the drilling spoils would be the result of mixing of the radioactivity on the concrete with the fill material above it which the drill displaces. The volume of material brought to the surface is calculated based on the borehole area and the depth of the basement floor. The drilling spoils are then spread at the surface. The licensee should perform a sensitivity analysis to understand the impact of assumptions on the distribution of residual radioactivity on the surface (area and thickness) to dose.

The second simulation would analyze the effect of leaching into the groundwater from the radioactivity on the concrete walls and floors. The concrete walls and floor can be assumed to not affect flow of radioactivity and the radioactivity on the floor can be modeled as a layer of contaminated soil at the depth of the floor. To simplify the simulation, the contamination on the walls may also be assumed to be on the floor. A reasonable contaminated zone thickness should be selected that will not underestimate the potential dose associated with the buried residual radioactivity and should consider whether there is surface or volumetric contamination of the concrete. The RESRAD model will produce a Dose-to-Source Ratio (DSR) in units of mrem/y per pCi/g for each radionuclide. The licensee should confirm whether surficial or volumetric residual radioactivity is present on building surfaces and should be able to measure and appropriately account for any residual radioactivity at depth to ensure that potential dose is not underestimated. If only surficial residual radioactivity is confirmed by radiological survey to be present and concrete surface DCGLs are needed to perform the final status survey, the licensee will need to convert the volumetric DCGLs to surface DCGLs. This conversion factor will require an assumed thickness to the contaminated zone, along with assumed densities of the fill and concrete material.

# APPENDIX K

# DOSE MODELING FOR PARTIAL SITE RELEASE

# K.1 Dose Modeling Considerations for Partial Site Release

This appendix consists of the technical guidance for reviewing the release of a portion of a site before final termination of the entire site, a process called partial site release (PSR). The review is carried out under 10 CFR Part 20, "Standards for Protection against Radiation," Subpart E, "Radiological Criteria for License Termination." This guidance is generally applicable to Decommissioning Groups 2–5.

The NRC has developed the guidance in this appendix to encompass the needs of the most complex situations, but the licensee should tailor the specific informational needs for a PSR request to the complexity and safety significance of the proposed action. This appendix is split into three sections. The first section, which complements Chapter 5 of this volume, details the review criteria to be used in assessing compliance with Subpart E. The second section provides technical information, which supplements the guidance in Appendices H and I. The third section contains two hypothetical simple examples of PSR considerations.

The guidance is focused on PSR requests that occur before the NRC approves the DP, but it is also applicable for requests for phased releases after DP approval (see Section K.1.8).

## K.1.1 Partial Site Release Reviews

For a PSR, dose modeling is not necessarily limited to the partial site but can also include residual radioactivity outside the partial site. Other areas on the site may contribute direct radiation or have natural processes that may move residual radioactivity to the area requested for release. For purposes of this section, "offsite sources" are potential sources of exposure that are not on the partial site but still on affected areas under (or previously under) the control of the licensee. For example, a licensee may have impacted groundwater under a portion of the site. At the time of the partial site request, the groundwater under the partial site may not be impacted. However, possible movement of this impacted groundwater (an "offsite source" of residual radioactivity) from the remaining site to the area requested to be released must be considered.

In addition to compliance analyses for the PSR, there should be evaluations of potential prospective analyses. These analyses should evaluate how the PSR could affect the license termination of the licensed site, including any additional PSRs. For example, releasing an area of the site at higher DCGLs than is likely for the rest of the site could constrain future decommissioning, forcing the licensee to use DCGLs for the rest of the site that are below what they could have been if the PSR had never occurred.

## K.1.2 Incorporation into Review Process

The licensee may use either a screening or a site-specific dose assessment to show compliance with Subpart E. Although it may use generic screening analyses to create the DCGLs of PSRs, the overall review should be a site-specific review. The NRC reviewer should use the appropriate section of Chapter 5 to review the assessments and the additional considerations for source terms, exposure scenarios, and pathway identification detailed below.

### K.1.3 Evaluation Information

The difference between a dose assessment for license termination of the entire site license and a dose assessment for a PSR is that other sources under control of the licensee may affect the potential dose on the PSR. In license termination of the entire site, when the site is released for unrestricted use, there are no offsite sources remaining under the control of the licensee to affect the projected dose for residents or workers using the site. In contrast, after a PSR, the remaining licensed site may still be operating and thus have dose contributions to the critical group receiving doses from the PSR, such as from surface water runoff or groundwater migration. In addition, sources on the remaining licensed site may result in doses to the public after unrestricted release of the remaining site in the future, such that members of the PSR critical group receive doses from sources on both the PSR and on the remaining (now terminated) site.

The NRC staff should review the licensee's assessment of offsite sources that may influence the dose analysis, and the evaluation of these sources would be like that of a source on the partial site. The development of the appropriate exposure scenarios for compliance evaluation should identify which sources NRC staff should focus on. The primary areas of additional consideration given to PSR cases in developing reasonable exposure scenarios are the following:

- How does the licensed site or a previous PSR influence the dose on the PSR (e.g., effluent releases, groundwater plumes, future combined use)?

- How could the PSR influence dose estimates for the licensed site during its decommissioning?

- How does the PSR influence previous PSRs (e.g., possible effects on the PSR's final DCGLs to limit the impacts on the previous PSR, so that the potential dose on the previous PSR does not exceed Subpart E criteria)?

### K.1.4 Development and Identification of Partial Site Release Exposure Scenarios

Based on the questions above, exposure scenarios can be divided into two categories: compliance and prospective. Analysis of both categories of exposure scenarios should assist in establishing the finality of the decision on the PSR.

Compliance exposure scenarios involve assessing the compliance of the proposed PSR, or the continued compliance of a previous PSR affected by the proposed PSR, with the Subpart E dose limit. Compliance exposure scenarios involve current or future exposure routes between the partial site and the previously released area or the licensed site (e.g., see the gamma radiation from the low-level waste storage area example in Section K.3.1). Compliance exposure scenarios that calculate exposures more than the regulatory limit or a licensee's self-imposed limit (e.g., from a previous PSR approval) should then entail remedial actions on the proposed PSR (not the previous PSR(s)) or more realistic dose assessments.

Prospective exposure scenarios involve assessing possible interactions between the partial site and any future decommissioning actions on the licensed site, including other PSR areas. The purpose of prospective analyses is to scope out the potential interactions in the future and address them, either by additional remediation of the partial site or by placing or acknowledging possible constraints on future decommissioning of the other sources.

### *K.1.4.1 Screening of Features, Events, and Processes*

The NRC staff should review the licensee's analysis using the worksheet in Appendix L of this volume to guide reviews of potential sources of interaction between the partial site and offsite sources. The purpose of this screening is to answer the questions from above, by identifying any potential interaction and evaluating the impact(s) on the dose calculations.

The licensee should have adequate justification for excluding each of the potential sources, transport processes, or exposure pathways not evaluated in the dose assessment analyses. Justification can be quantitative or qualitative.

Three methods are acceptable for handling the impact of offsite sources related to interactions that have not been screened out: (1) incorporate the source, transport mechanisms, and pathways into the conceptual model and the dose analyses, (2) remediate those sources, or (3) apply constraints on the PSR's DCGLs, to accommodate potential exposures from offsite sources or from previous PSRs.

Therefore, the NRC staff should verify the following:

- The licensee screened potential interactions with the licensed site and previous PSRs.
- The screening arguments are justified.
- The licensee properly addressed the remaining potential exposure pathways.

Section K.3.1 illustrates some of these considerations.

### *K.1.4.2 Screening the Use of the Partial Site and Other Areas by the Critical Group*

A member of the critical group could potentially be exposed to higher doses than those resulting from the PSR alone. This would be through the use of other impacted areas after they have been released (including previous PSR areas), in addition to continuing the use of the partial site.

Three general situations can result in doses to individuals that are higher than for the PSR area alone:

- One of the land area's DCGLs took into account the small size of the area.
- Use of more than one exposure area would result in the dose receptor receiving exposure from radionuclides or sources not present on the released area.
- Use of more than one exposure area would result in the dose receptor receiving exposure from new exposure pathways or would increase the degree of exposure to a current exposure pathway.

Section K.3.2 illustrates a hypothetical review of a situation involving multiple land uses.

If the licensee has used the same DCGLs for a previous PSR or commits to use the same DCGLs for areas surrounding the partial site, multiple uses of the areas is not likely to result in a

higher dose, as long as none of the above situations is present and the exposure scenarios and assumptions used in the calculations are appropriate for all areas.

If the licensee has: (1) used different DCGLs; (2) has at least one of the above situations present; (3) found that the exposure scenarios and assumptions on the proposed PSR used for a previous PSR are no longer appropriate; or (4) has not committed to using the same analyses for surrounding areas (as long as it would be valid for the other areas, as well), then the NRC staff should evaluate the licensee's analyses of potential multiple use exposure scenarios. For example, for interactions with a previous PSR, the staff should review: (1) any prospective analyses and associated constraints, if established, on previous PSR(s), (2) the estimated dose from the residual radioactivity related to both the previous PSR and the proposed PSR, and (3) any new or updated analyses performed by the licensee.

### K.1.5 Partial Site Release Evaluation Criteria

The NRC staff should verify the following PSR considerations:

- For PSR and previous PSR interactions:
  - The exposure scenarios used in the prospective analyses for the previous PSR, that analyzed the interactions between the previous PSR and the area encompassed by the proposed PSR, continue to be appropriate, or have been updated appropriately.
  - The licensee incorporated any constraints imposed by the previous PSR that remain appropriate in determining the DCGLs for the proposed PSR.
  - The licensee appropriately identified those sources that may affect the dose to the average member of the critical group, from either the previous PSR or the proposed PSR.
  - The licensee adequately justified each excluded potential source, transport mechanism, and pathway.
  - The licensee incorporated, or addressed by other appropriate means, any sources, transport mechanisms, or pathways that could not be eliminated.
  - The licensee evaluated (either quantitatively or qualitatively) reasonable exposure scenarios to account for interactions between the previous PSR and the proposed PSR. This includes the prospective analyses for the previous PSR, as well as any new exposure scenarios that required evaluation based on new information.
  - The DCGLs for the proposed PSR should not result in exposures exceeding the dose limit for either the previous PSR or the proposed PSR. The dose assessment for the proposed PSR should also include any appropriate contributions from the licensed site.
- For the PSR and interactions with the licensed site, considering both current and future sources (e.g., potential impacts from other decommissioning activities, or potential future parallel use of impacted areas on the licensed site and the PSR):

- The licensee appropriately identified those current and potential future offsite sources that may affect the dose calculated for the partial site.

- The licensee adequately justified each excluded potential source, transport mechanism, and exposure pathway.

- The licensee incorporated, or addressed by other appropriate means, any sources, transport mechanisms, and exposure pathways that could not be eliminated.

- The licensee evaluated reasonable exposure scenarios to account for interactions between the proposed PSR and the licensed site, including any prospective analyses that estimate exposures after the licensed site is decommissioned.

- The licensee ensured that the DCGLs would not result in exposures exceeding the dose limit at the proposed PSR. The dose assessment for the proposed PSR should also include any appropriate contributions from previous PSRs.

- The licensee has clearly documented any constraints placed on current and potential future sources of exposure on the licensed site.

### K.1.6 Dose Modeling Approaches

Licensees proposing PSRs may still be able to use either dose modeling option: (i) screening values or (ii) site-specific analyses.

- If a licensee proposes to use the screening criteria, the NRC should ensure the following:

  - Interactions with the licensed site or previous PSRs have been appropriately evaluated.

  - Any sources of potential exposure from the licensed site have been either constrained or remediated.

  - Any sources of potential exposure increasing either the dose to residents or workers from the proposed PSR or a previous PSR have been either constrained or remediated.

  - The screening criteria have been appropriately scaled by all the considerations associated with the PSR. For example, in Section K.3.1, the licensee limited the groundwater dose to 0.05 mSv (5 mrem). Therefore, the screening values for the PSR's surface soil would need to be scaled to 80 percent (0.2 mSv (20 mrem)/0.25 mSv (25 mrem)) of the published values or those received by using the current version of the DandD computer code.

  - The PSR and its analysis meet the other requirements of Sections 5.1.1 and 5.1.2, as appropriate.

- If a licensee uses site-specific modeling, the NRC staff must verify the following:

- The licensee has incorporated all sources from the licensed site or previous PSRs, as necessary, into the analyses.
- The licensee has properly reflected any constraints in calculating the DCGLs.
- The modeling meets all the other review criteria of Section 5.2, as appropriate.

### K.1.7 License Termination: The Effect of Previous Partial Site Releases

At the time of final license termination, the NRC staff should take into consideration any previous PSRs. The entire site (including the previous PSRs) should meet the Subpart E dose limit. Reviewing the impact on the license termination is the same as that discussed under the first item in K.1.5, "For PSR and previous PSR interactions." In this case, it is necessary to consider the rest of the licensed site as the PSR.

### K.1.8 Use of Partial Site Release During Decommissioning

Reviewers can use this guidance when licensees request the release of portions of their site(s), either as part of a DP submittal or after the DP has been approved. After the DP has been approved, some of the issues are not as relevant. If the licensee has prepared a DP for the entire site, more information will likely be available at the time of the PSR. Importantly, the NRC reviewer may be able to review the PSR's DCGLs, the DCGLs for other areas of the site, and any plans on continued remediation of other areas of the site. The licensee may still need to complete the prospective analyses of critical group behavior after the entire site is released, but these exposure scenarios are likely to be easier to define and evaluate.

## K.2 Partial Site Release Technical Basis for Dose Modeling

### K.2.1 Considerations for Partial Site Release Dose Assessments

Although the license termination requirements in Subpart E provide options for unrestricted and restricted release, PSRs would normally be used for unrestricted release. A PSR has many aspects in common with the existing approach for unrestricted release, and the available guidance is generally applicable. One key difference is that the PSR does not occur concurrent with license termination. As a result, continuation of licensed activities outside of the released area represents a potential source of exposure. In turn, the residual radioactivity from the PSR may affect dose analyses for other areas of the facility during subsequent PSR requests or eventual license termination.

Because this volume's guidance requires that the dose assessment include all significant exposure pathways, the need to consider the potential for accumulation processes resulting in increased radionuclide concentrations over time is not a new concept for PSR. Nonetheless, the importance of accumulation is increased under a PSR because the license will not be terminated. The existing site areas outside the partial site are not required to be remediated at the time of the PSR, thereby increasing the potential for accumulation on the partial site, which could affect the dose assessment.

One of the most important concepts of the guidance for a PSR is the finality of the decision. The purpose of the guidance is to establish the scope of the review and focus the reviewer's attention and resources on early identification of aspects important to compliance. The primary objective of the PSR guidance is to ensure that any PSR meets Subpart E requirements, even if

potentially affected by later PSRs or license termination. The secondary objective of the PSR guidance is designed to ensure, at the time of license termination, that all prior released areas are considered and included, as necessary, in dose assessments to provide assurance that the entire site meets Subpart E requirements. To meet these two objectives, the licensee is asked to perform both compliance calculations for current conditions at the PSR (or the effect on a previous PSR) and prospective calculations to estimate the impact on other decommissioning activities by a licensee. This set of analyses should help ensure that the DCGLs chosen for the PSR do not result in the need for future remediation of the PSR or unduly constrain the decommissioning of the entire site.

The existence of a PSR may place constraints on future activities that occur nearby, including currently licensed activities to limit the potential for exposures to a critical group residing or working (nonradiation workers) on the released area from exceeding other public dose limits (see Section 3.4 of this volume). For example, 10 CFR 20.1301(d) requires that a fuel cycle facility comply with the requirements of 40 CFR Part 190, "Environmental Radiation Protection Standards for Nuclear Power Operations." This standard (40 CFR Part 190) limits the total dose that a member of the public may receive from all fuel cycle activities. For the remaining site to show compliance with 40 CFR Part 190, the dose from the partial site and any effluents from the remaining site's operations will need to be combined. The constraint, then, is not on the compliance of the PSR with Subpart E limits but on maintaining compliance with 40 CFR Part 190 for the remaining activities (e.g., an independent spent fuel storage installation). Existing NRC effluent control and operational dose limits and their associated guidance should generally limit operational releases to acceptable levels. Adjustments may need to be made to effluent compliance calculations or environmental sampling areas to account for the removal of the PSR from licensee's control (e.g., because of changing the site boundary). The NRC may develop additional guidance addressing these constraints (other than 10 CFR Part 20, Subpart E) in the future.

The dose criterion of 40 CFR Part 190 is based on actual annual doses. Thus, it should be noted that a licensee does not have to use its Subpart E compliance calculation (generally a prospective calculation) as the dose contribution from the PSR for the compliance determination under 40 CFR Part 190. Information will be available to the licensee to estimate the actual annual dose based on the actual activities that occurred at the released site. For example, a partial site is released using an exposure scenario similar to a residence. The next year the licensee is performing calculations to show compliance with 40 CFR Part 190. In the year since the PSR was approved, the land was used as a public parking lot. For compliance with 40 CFR Part 190, the licensee can evaluate the PSR dose to a member of the public using the parking lot instead of the residence exposure scenario used in approving the PSR for 10 CFR Part 20, Subpart E. The licensee may also account for decay since the FSS.

For nonimpacted areas, the NRC technical review should normally be limited to the sufficiency of bases in the site characterization. A PSR, effectively, narrows the definition of nonimpacted, because of the possibility of future licensee actions resulting in an effect on the PSR. For example, in some cases, close proximity to existing operations, contaminated areas, future remediation sites, or potential storage areas may not allow a licensee to designate an area as nonimpacted and release it without a dose assessment. For impacted areas, PSR requests can involve a more complicated compliance demonstration and review effort. The site characterization should include areas of the site outside of the PSR to the extent, necessary to provide assurance that residual radioactivity from the licensed site (or a previous PSR) is unlikely to transport material to the partial site that would result in potential future exposures to users of the property (including subsurface water and groundwater).

### K.2.2 Partial Site Release and Decommissioning Guidance

This NUREG report was written, in large part, to address decommissioning of sites as part of the license termination process. Consequently, it often discusses termination of the license as the end result of decommissioning. When applying this guidance to a PSR, most of the references to license termination should be regarded as implying the completion of the partial site decommissioning effort. Despite the frequent use of the term "license termination," licensees should be aware that a PSR will not result in license termination, as the entire licensed site (including any PSRs) should meet the Subpart E requirements at the time of license termination. Therefore, true license termination issues only need to be considered in PSR reviews when assessing prospective analyses that may raise issues that need to be considered or analyzed at the time of license termination (e.g., creation of new license conditions that identify pathways that should be included in DCGL calculations). Licensees should also be aware that the existence of a PSR adjacent to impacted areas could place limitations on future decommissioning methods and actions related to the license termination (e.g., to minimize the potential for decommissioning to re-contaminate previously released areas).

This NUREG report uses the terms "site" and "facility" interchangeably. Most of the references will apply to the boundaries of the area proposed for a PSR (i.e., the area to be decommissioned). Exceptions to this are when the consolidated guidance discusses the need to collect site characterization information, in which case the terms "site" or "facility" can include areas beyond the boundary of the PSR and potentially encompass the entire site and any previous PSR(s), as necessary to establish contaminant source, transport, and exposure pathways for DCGL calculations. To resolve such safety concerns, the NRC staff is expected to use pre-submittal meetings with the licensee to develop the information needed on specific areas of the licensed site.

### K.2.3 MARSSIM and Partial Site Release

The MARSSIM approach involves demonstrating compliance on a survey unit-by-survey unit basis. Survey units are determined based on the expected level of residual radioactivity in areas across the site, as well as spatial and topographical considerations. By allowing compliance demonstration by survey unit, the current approach is congruent with a PSR concept. As a result, in general, the licensee can apply the MARSSIM approach directly to a PSR without significant problems.

To limit the potential for interactive dose effects, the licensee should fully include in the proposed PSR final surveys any impacted areas that continue across the proposed PSR boundary. If buildings are intended for PSR, the licensee should include them unless it can demonstrate a low potential for future exposure of individuals in the PSR portions of the building from other impacted areas of the building and should include any significant dose contributions from outside the PSR area in determining the DCGLs.

### K.2.4 Dose Modeling Specific Issues

The compliance methodology in this NUREG report emphasizes dose modeling to derive DCGLs to be used as input to the MARSSIM process (see Section 2.5 of this volume). Simple sites that only involve surface contamination and a low potential for migration of residual radioactivity should require straightforward dose calculations to derive DCGLs. Sites with both subsurface and groundwater residual radioactivity or migration of radioactive material from one

area to another generally require more complex modeling and compliance demonstration methods.

A primary concern is that the dose calculation of DCGLs include all applicable transport and exposure pathways, because other parts of the site may not be remediated at the same time as the PSR area. The licensee should give special consideration to any potential for significant transport of material into the released area from outside the boundary, or from the released area to other areas of the licensed site or to another previously released area. DCGLs should account for the movement of radioactive material where accumulation processes could lead to significantly increasing media concentrations if the transport were included or would add new radionuclides or exposure pathways to the PSR dose assessment.

For example, an area designated for PSR may not have impacted groundwater, but an impacted area on the licensed site up-gradient has impacted groundwater that is expected to migrate into the PSR in the future. The licensee should include future groundwater residual radioactivity in the dose calculation for the PSR (unless its contribution would not be significant) or address it by other methods. The licensee may need to limit surface DCGLs for the PSR to ensure that the total PSR area, including any future groundwater dose, complies with the 0.25 mSv/y (25 mrem/y) dose limit for the 1,000-year compliance period. Similar situations could exist with up-gradient surface or subsurface contamination (e.g., leaching and transport from the sources to the PSR area).

Similarly, residual radioactivity sources within the PSR area should be evaluated for potential transport to the licensed site or other previous PSR areas. Most licensees should be able to assess the potential for transport pathway communication between site areas using available site characterization information. Complex sites may require the collection of additional site characterization information (inside and outside of the PSR area) to support the evaluation of transport pathways. The scope of site characterization work should be consistent with the expected level of residual radioactivity and potential dose consequences.

The PSR process requires records of the PSR residual radioactivity to ensure inclusion in subsequent PSR analyses and in the overall site license termination process. Residual radioactivity in PSR areas may be a concern for site license termination when the potential exists for the migration and accumulation of radioactive materials to the licensed site. This circumstance may only be significant when several PSR areas exist, in close proximity, and share common transport pathways with the licensed site, allowing the potential accumulation of transported material. Another situation arises when the critical group may use multiple PSR areas. In addition, the NRC's approval of the PSR may have required the licensee to agree to limits on the dose contributions from decommissioning activities or residual radioactivity sources remaining on the licensed site.

For each subsequent PSR request and at license termination, the licensee should consider all prior PSR areas for potential contributions to dose when calculating partial or site DCGLs. For some sites, this could mean that prior PSRs could constrain the amount of residual radioactivity allowed in a future PSR request or at the remainder of a site at license termination. If the review of a PSR identifies important features, events, or processes (FEPs) to be considered at license termination, the NRC staff may impose a license condition to ensure the licensee addresses the matter.

The review of impacts on previous PSR areas is very important because the NRC approved the previous PSR request based on the calculations and evaluations that showed compliance with

the Subpart E limit. If another portion of the site is decommissioned and released later, the NRC should review the possible impacts on the dose estimates at the previously released site. The NRC should first review the prospective analyses from its previous approval. If they remain valid and bound any impacts that the proposed PSR could cause, then it can consider the impact of the proposed PSR to be acceptable. The NRC should ensure that the DCGLs for the proposed PSR prevent the dose from previous PSR areas from exceeding the Subpart E limit.

Although exposure scenario development, especially for prospective analyses, does involve some speculation, both human behavior and FEPs should be based on present knowledge. To limit the degree of speculation, the NRC staff should focus on reasonable exposure scenarios and avoid speculation on activities that are not present in the region, are not reasonably likely to occur, or that would change human behavior or the FEPs. For example, a scenario that involves modifying the local topography to allow surface water to transport radioactive material from an impacted area to the PSR area is generally too speculative and not a reasonable scenario unless it is part of a remediation option.

An important part of the detailed technical review may be determining if a licensee has included all applicable exposure pathways in the DCGL calculations and provided sufficient bases for exclusion of exposure pathways. Applicable exposure pathways are determined by considering all three of the following:

(1) how the critical group can be exposed to localized residual radioactivity

(2) the potential for sources and transport of radioactive materials (from the PSR area, the licensed site, or previous PSR areas) to the location of the applicable critical group

(3) concurrent use, if appropriate, of the PSR area and previous PSR areas by the critical group

This NUREG report generally addresses exposure pathways for localized residual radioactivity, and the methods are relatively straightforward. This section focuses on analyzing sources and transport pathways, because the potential risk of additional sources of exposure affecting a PSR (or a previous PSR) increases when the entire site is not decommissioned at the same time. Exposure scenario definition and pathway identification are therefore key aspects of DCGL dose modeling that are affected by the unique circumstances of a PSR.

### K.2.5 Features, Events, and Processes

Applicable source, transport, and exposure pathways comprise the exposure scenario for DCGL calculations. The licensee can calculate DCGLs using the all-pathway models, or pathways can be decoupled from the modeling and their results allocated to pathway-specific DCGLs that can be combined to generate a survey unit or PSR $DCGL_W$ that equates to the Subpart E dose limit. The licensee may use the site conditions, complexity, and level of risk to choose a specific method among several options for calculating DCGLs.

#### *K.2.5.1 Screening Methods*

The purpose of screening various sources, transport mechanisms, and exposure pathways is to evaluate whether the PSR may have processes that could result in radioactive material being transferred between the PSR area and either the licensed site or previous PSR areas. The first goal of the screening criteria is to eliminate various FEPs from consideration while minimizing

the amount of information the NRC staff needs to make a decision. A second goal is for the screening criteria to factor in the availability and cost of obtaining the information (i.e., the first criterion should not require the development of a complex site-specific, three-dimensional groundwater model). The screening of these criteria should not only focus on the effect of the licensed site, or a previous PSR, on the PSR, but, also, the potential contribution of the PSR on the dose assessment for the entire site at the time of decommissioning, or the current compliance of a previous PSR with Subpart E. The following are the general categories of screening criteria:

- the presence of residual radioactivity in various media, including effluent releases from the operating site (e.g., soil, groundwater, air)

- the availability of mechanisms to either move material from one location to another, (e.g., groundwater movement) or project exposure from one area to another (e.g., direct radiation)

- the availability of exposure pathways to cause a dose in humans after material is moved or projected to the area

The licensee may screen out a medium, such as groundwater, that is found to contain residual radioactivity if it has minimal levels of residual radioactivity (compared with the residual radioactivity currently present in the media at the critical group location). If the source remains, then the licensee screens the transport mechanism(s) to evaluate the capability of the process to move material to the area of interest. This result can then be compared to the residual radioactivity levels for each radionuclide currently present in the area of interest or other processes for moving material. Finally, the licensee can screen the potential exposure pathways to remove those that would result in insignificant doses or are not present at the location where the material is being deposited.

In formulating a complete exposure scenario for a proposed PSR, the licensee should give initial consideration to available information (from the PSR area, the site, or any prior PSR) that can rule out further consideration of specific sources or transport pathways. Although investigation of potential sources and transport pathways can become complicated, the licensee can rule out potential sources and transport pathways with relatively simple and available information. Appendix L provides a worksheet of source, transport, and exposure pathways with questions that can be used for screening. Use of a “top-down” approach to screening can avoid an unnecessary and costly investigation into details that may not have a significant impact on DCGL calculations. It is expected that, once a potential release or transport pathway has been identified, licensees may provide simple, yet reasonably conservative, screening-type calculations to assess importance. Licensees may exclude pathways that make only a small dose contribution, if the pathway results in less than 10 percent of the dose limit, and the sum total of all pathway exclusions does not exceed 10 percent of the dose limit (see Section 3.3 of this volume). The licensee should clearly identify all screened pathways and should show sufficient bases for exclusion.

**Example of Screening Process for Features, Events, and Processes**

In Appendix L, a worksheet has been provided as one method of screening FEPs for a PSR. The purpose of the worksheet is to provide some general topics that can, in most cases, be considered, with generally available information, to minimize unnecessary site characterization,

modeling, and review. The licensee can use the worksheet to develop both compliance and prospective analyses. Ultimately, if radionuclides cannot be released or transported to the critical group location, there is no point in including the FEP(s) in the dose assessment.

Site-specific conditions and available information may make it desirable for a licensee to initially focus on source, transport, or both, when trying to screen FEPs. In some cases, it may be necessary to conduct limited dose calculations to justify the exclusion of a source, transport mechanism, or exposure pathway. If the licensee cannot screen out a source, then it should consider it for transport screening. If pathways cannot be excluded using this worksheet, the licensee should consider them in initial dose calculations, either by including them in the analysis or by modifying the dose limit using an agreed-upon limit. Results of the initial dose calculations can provide additional insights to the significance of pathways with respect to dose and may offer additional means for further refinement of the calculations to address only the important features and processes. The licensee should identify all source, transport, and exposure pathway exclusions from modeling and appropriately justify their exclusion.

The worksheet is split into three parts: (1) sources, (2) transport processes, and (3) exposure pathways. The method is to start with the source questions and follow the directions under each item, as necessary. The user should follow the path down until the item is screened out or needs to be considered in the analyses. After reaching the end of a path, the user should go back to where the branching occurred and continue with the questions, if applicable. For example, a site has some residual radioactivity in the soil, and the licensee reviews the questions under L.2.2.2 (“Soil Transport: Leaching”). The questions lead the user to L.2.4 (“Groundwater”) and the user follows that path to its conclusion. The user then continues to evaluate that residual radioactivity in soil by returning to questions L.2.2.3–L.2.2.6.

In general, for each “yes” answer, the analysis continues to more detailed questions on that source and media type. Each “no” answer on a black bulleted question means that area (and its related questions) need no further evaluation for that specific source or media combination. For a black-bulleted question with a list of more detailed questions (i.e., with the dash bullets) to be excluded, all of the detailed questions need to have “no” answers. Some instructions may provide exceptions to this general rule.

For example, the last question of L.2.3.1 (“Deep Soil Transport: Leaching”) includes three specific transport mechanisms from deep soil. If the answers to all three were “no,” the leaching of the source would be screened out of the dose assessment. If the answer were “no” for surface water and other, but “yes” for groundwater, the user should address potential leaching of the deep soil source unless the groundwater transport or related exposure pathways were subsequently screened out.

### *K.2.5.2 Human-Induced Exposure Scenarios*

Another source of exposure that may lead to interactive dose effects between the PSR and another impacted area under (or previously under) control of the licensees occurs when individuals using both the PSR area and the other impacted area(s) after the licensee no longer controls those areas. The concern is that a critical group could use the PSR area, such that it still receives a large fraction of the Subpart E dose limit, and reasonably use another impacted area that would lead to the critical group receiving, in total, doses in excess of the Subpart E limit.

Most prospective exposure scenarios evaluate the human-induced exposure scenarios after the other impacted area is released for unrestricted use. In cases where the human-induced exposure scenario involves a previous PSR, the analysis is one of compliance for the PSR, which could also verify that the human-induced exposure scenario may not result in exposures to the previous PSR, above the limit. The licensee can use self-identified and agreed-upon limits to address the exposures from the future use of areas that are on the licensed site (see the example in Section K.3.2).

Three situations can result in dose assessments higher than those for the PSR alone:

(1) One of the DCGLs for the land areas considered the small size of the area.

(2) Use of more than one impacted area would result in the dose receptor receiving exposure from radionuclides not present in the PSR area.

(3) Use of more than one impacted area would result in the dose receptor receiving exposure from new exposure pathways or would increase the degree of exposure to a current exposure pathway.

Taking an area's size into account when developing the DCGLs is a special case of the third bullet above. This is because size-related modifications for dose modeling usually result in reducing the number of pathways or amount of exposure, but these changes may not be obvious, especially if the code itself (like RESRAD-ONSITE) modifies the dose calculations.

### K.2.6 Subsurface Residual Radioactivity

Subsurface residual radioactivity can exist in soils and deeper geologic strata. Common sources of subsurface residual radioactivity include material leached from surface soils, buried waste, and impacted groundwater. Impacted areas can be either saturated with groundwater or unsaturated (where water may percolate through but does not fill all pore spaces). Currently, this volume suggests applying the MARSSIM (surface-based) methodology to subsurface radioactivity, with a few modifications to address volume sources (see Appendix G, Section G.2.1). The NRC expects to update its guidance in the future to improve statistical methods for subsurface residual radioactivity. This section discusses special considerations for addressing subsurface residual radioactivity under the PSR exposure scenario(s), with an emphasis on pathway identification for DCGL calculations. Because addressing subsurface residual radioactivity is merely a component of the same dose modeling discussed in the previous sections, the same framework for DCGL calculations applies. For the purpose of discussion, surface water is included in some examples because of the interconnection between the surface water and groundwater systems.

If the MARSSIM methodology classifies a site as impacted, this volume suggests that surface water and groundwater surveys be designed on a site-specific basis. If the licensee does not have important information necessary to understand subsurface characteristics (including extent and amount/type of residual radioactivity) immediately available when requesting a PSR, some characterization of surface water flow, sediment movement, and groundwater flow for both the PSR and adjacent areas, as necessary, on the licensed site may be required to support the amendment request. The source locations, in conjunction with the site complexity, determine the surface and groundwater characterization needed at the time of a PSR request. The level of surveys for surface and groundwater residual radioactivity should factor in all three of the following:

(1) the extent of existing residual radioactivity of soil on the PSR

(2) the proximity of the PSR to existing and potential impacted areas on the licensed site

(3) the complexity of the surface and groundwater hydrology

As noted previously, a PSR that has been classified as impacted requires dose modeling. The licensee should assess subsurface residual radioactivity, once identified, for inclusion or exclusion in dose modeling to derive DCGLs. Residual subsurface radioactivity that contributes less than 10 percent of the dose limit does not need to be included in the DCGL calculations, as long as all exclusions do not consist of more than 10 percent of the dose limit, but its exclusion should be documented for future consideration at license termination.

Simple situations that need to include subsurface residual radioactivity in dose modeling may involve only radioactive material originating from the PSR area or only one offsite source of impacted groundwater in a relatively simple hydrology system. More complex PSR scenarios can involve numerous additional sources of residual radioactivity migrating from outside the PSR area or migrating from the PSR area onto a previous PSR area or the licensed site. An important aspect is the possibility of multiple sources coalescing in the surface or groundwater systems (i.e., the additive effect of multiple sources from the licensed site, the PSR area, or previous PSR areas).

Licensees should consider all potential processes for the migration of material; however, they can easily exclude some pathways with available information (see Section 3 of Appendix I and Appendix L). There is a large dilution effect when radionuclides migrate into bodies of water, such as streams, rivers, lakes, and ponds. Sediment movement and groundwater flow are commonly slow processes, relative to surface water flow. Reduction of residual concentrations in groundwater (caused by mechanical mixing and sorption) and radioactive decay effects are associated with the longer time factor in the transport legs for sediment movement and groundwater transport. A clear example of pathway exclusion would be a PSR area located in a watershed that is isolated from the licensed site operations and impacted areas and is located upstream of all other offsite sources. It is reasonable, in this case, to neglect the residual radioactivity at the PSR area in dose modeling at the time of license termination if it can be demonstrated that there is no significant dose contribution to the site DCGLs. Another simple example is the exclusion of drinking water pathways given the absence of a drinking water aquifer accessible to the critical group.

Types of surface and groundwater features that could lead to focusing residual radioactivity from multiple, spatially separated source areas are separated into two categories—common features and site-specific features. To determine if focusing occurs, site characterization data identify spatially convergent groundwater flow directions or convergent surface water flow and sediment movement. The license should determine the level of site characterization by the potential for these features to occur at the site. Examples of each are described below.

The most common feature leading to convergent mass movement is a river, stream, or pond in a watershed. Multiple radionuclide sources at various locations around a watershed could all potentially migrate in the surface and subsurface towards the main stream channel or pond. All surface water in the watershed could be routed into the main channel or pond. Whereas most watersheds have an outlet, some lakes, ponds, or bogs may be the terminal point in a transport pathway where residual radioactivity may accumulate. Changing chemistry of the transport path

(e.g., the reducing environment of a swamp) can also affect the deposition or dissolution or mobilization of specific contaminants.

In that same watershed, the uppermost aquifer may also focus groundwater flow into the stream, because gradients in the unconfined aquifers typically follow the topography and commonly seep into stream and river channels. The exception applies to uppermost aquifers with water tables that lie below the stream elevation; these aquifers would not necessarily seep into the stream channels or ponds and would not lead to a convergence of groundwater flow directions unless dictated by another feature.

Site-specific features, such as faults, karst terrains, and alluvial channel deposits, can focus water from diverse locations into single transport pathways. These features may lead to a channelization of flow in the subsurface. Licensees should first determine if such subsurface features exist at a site. If so, they can analyze the candidates for the potential to focus transport pathways from impacted areas of the licensed site or a previous PSR area.

Facilities and PSRs with the potential for multiple sources of residual radioactivity that could migrate to surface or groundwater should use or obtain sufficient site characterization data to show the possibility for convergent features to exist on the site. The licensee may have to get the site characterization data at the time of a PSR request if they are not already available. The licensee should assess the potential for overlapping transport pathways from multiple source areas, where those sources could be at the PSR area, the licensed site, or previous PSR areas.

### K.2.7 Records and Documentation

Maintaining complete records of PSRs is important, because the information is likely to be needed for any subsequent PSR requests and at the time of license termination. The NRC staff should consider all prior licensing actions in the reviews for a license termination, of which a PSR is only one example. Similarly, the framework for PSR involves considering all prior PSRs and whether the residual radioactivity needs to be included in DCGL calculations for license termination. Because considerable time may elapse between a PSR and the eventual license termination, maintenance of complete records is an important aid to the licensee, as well as the NRC staff. Incomplete records may result in the need for additional site characterization at the time of license termination. Records should include identification of impacted areas and information describing the MARSSIM RSSI methods used and results obtained, including all PSR site characterization information that supports DCGL calculations. Any information supporting source, transport, or exposure pathway exclusions in the PSR area in common with the licensed site is important, as are any licensee agreed-upon limits used to simplify the previous dose assessments. This information may support a determination of whether PSRs must be included in DCGL calculations at the time of license termination. Volume 3 of this NUREG has more information on recordkeeping requirements.

## K.3 Hypothetical Examples

### K.3.1 Contributions from the Remaining Licensed Site

A licensee requests the release a portion of a site 10 years before the date it estimates it will provide the NRC with the DP. The PSR has surface soil residual radioactivity of cobalt-60 (Co-60) and cesium-137 (Cs-137). Adjacent to the PSR, on the licensed site, is the low-level waste storage area, which is a potential source of gamma exposure to individuals within the released area. The only other potential offsite source is a groundwater plume from the licensed

site. The licensee evaluates the two offsite sources and eliminates all other offsite sources because of the absence of valid transport mechanisms to allow significant impact on dose analyses. The licensee then takes the two following actions to address the remaining potential exposure sources:

- A berm is going to be built between the low-level waste storage area and the PSR area, on the licensed site, to reduce the external gamma exposure. The berm is estimated to reduce the potential dose from 0.05 mSv (5 mrem) to less than 0.001 mSv (0.1 mrem). The contribution is now insignificant, and the source can be eliminated from further consideration in estimating the dose for the compliance calculations for the PSR's DCGLs. The presence of the berm would then likely become a license condition. Removal of the berm in the future may require reanalyzing the total dose to the critical group to verify that exposures from the released area will not exceed the Subpart E limit after unrestricted release of the low-level waste storage area.

  **Note that, if a previous PSR is impacted by a proposed PSR, or the decommissioning of the site, such that doses from the PSR area may exceed the Subpart E dose limit, constraints are to be placed on the current action(s) and do not require that the previous PSR be remediated (except as noted by 10 CFR 20.1401(c)).**

- The groundwater plume is estimated to reach the PSR area in approximately 15 years. The licensee currently has no final plans to remediate the plume. Current conservative dose modeling estimates the annual peak exposure to be approximately 0.05 mSv (5 mrem) from an all-pathway analysis, using the groundwater concentration at its current location. The licensee proposes partitioning the unrestricted dose limit for the PSR and constraining the annual peak dose from the surficial soil residual radioactivity in the PSR area to 0.2 mSv (20 mrem) and the groundwater dose to 0.05 mSv (5 mrem). To ensure the groundwater concentrations do not exceed specific concentrations associated with the agreed-upon limit of 0.05-mSv (5-mrem), the licensee may install monitoring wells between the plume and the new licensed site boundary and develop a corrective action plan to use in case the concentrations rise above some specified fraction of the licensee's agreed-upon limit.

Although the PSR should meet the Subpart E dose limits at the time of approval, these actions may further affect the final decommissioning of the licensed site. For example, assume, at the time of license termination, residual amounts of Co-60 and niobium-94 are in the surface soil around the low-level waste storage area and a building code requires the berm to be removed before the entire site is released for unrestricted use. When the berm is removed, external exposure would result in 0.04 mSv/y (4 mrem/y) to the average member of the critical group in the PSR area. Based on the concentrations from the PSR's FSSR and the groundwater agreed-upon limit, the total dose estimate for the PSR is now 0.29 mSv/y (29 mrem/y), which does not meet the Subpart E dose limit for unrestricted release. The effect of the PSR may limit the final DCGLs of the low-level waste storage area. Other options available to the licensee would be to reevaluate the PSR dose assessment to account for decay and new information on the dose from the groundwater plume or provide additional remediation of the groundwater plume.

### K.3.2 Use of Multiple Areas

A second licensee requests the release of an impacted portion of a licensed site. The area has residual radioactivity of uranium and thorium. The PSR area is rocky, with poor soil, and the licensee can provide adequate justification that the critical group would not plant extensive gardens nor use the groundwater under the released site. No offsite sources or transport mechanisms could affect the dose if the critical group used only the PSR area.
The licensed site has the closest other radioactive source, under the control of the licensee—some groundwater concentrations of hydrogen-3 and chlorine-36 from old tracer tests. The land over the groundwater residual radioactivity is suitable for extensive gardening or farming, and the aquifer is potable. As part of the dose assessment, the licensee evaluates a prospective exposure scenario where the critical group may use both the PSR area and portions of the licensed site after it is decommissioned. After review of the sources, impacted areas, and routes of exposure, the licensee decides that a reasonable exposure scenario would involve the person living on the previously released site and using the offsite area and its impacted aquifer for drinking water and growing an extensive garden. The licensee, believing it should be easy to remediate the groundwater, addresses the offsite pathways in this exposure scenario by proposing an aggressive limit (i.e., a small fraction of the current dose estimate) on the dose from the waterborne pathways (which are all from the offsite area). Accounting for its agreed-upon groundwater limit, the licensee calculates DCGLs for the PSR based only on the radioactivity in the PSR area, performs the FSS, and gains NRC approval of the PSR request for unrestricted use.

At the time of site decommissioning, years later, the licensee, having better characterized the groundwater plume and having run some well pumping tests, finds that it may be difficult to meet the limit it established on the groundwater dose without extensive remediation of the groundwater. Therefore, the NRC staff's perspective is that at license termination, the licensee is effectively left with three options:

(1) Remediate the groundwater down to the licensee's agreed-upon limit.

(2) Revise the licensee's agreed-upon limit based on additional modeling (e.g., considering actual FSS results for the PSR, decay of the sources, new information known about the groundwater system, and associated residual radioactivity, or more realistic models of groundwater dispersion and transport).

(3) Combine the two options.

The NRC views remediating a previous PSR as the option of last resort, consistent with its policy on intervention of terminated licenses. Obviously, if the licensee desires to remediate the previous PSR area, the NRC would not necessarily stop the licensee. However, the situation could involve several issues related to regulatory authority and require the current site owner's approval.

**Note that if the groundwater with residual radioactivity had been under a previous PSR instead of the licensed site, the options would have been different. The options would be to remediate the uranium and thorium residual radioactivity in the proposed PSR area, do more complex modeling, or combine the two.**

# APPENDIX L

# WORKSHEET FOR IDENTIFYING POTENTIAL PATHWAYS FOR PARTIAL SITE RELEASE

## L.1 Introduction

This volume provides a worksheet to assist the staff of the NRC and its licensee to screen potential sources and transport pathways considered in dose modeling for derived concentration guideline levels for a PSR. It is intended that the results of this worksheet summarize the exclusion or inclusion of each item and the screening argument, as well as reference the more complete screening argument, if necessary. The questions should be considered for all sources of residual radioactivity and potential critical group locations. Although this worksheet has been designed for use in identifying features, events, or processes that could result in additional sources of exposure for the critical group in the partial release, it can also be used for general exposure scenarios and pathway development.

**The worksheet focuses on physical features, events, and processes that may transport radioactive material to the partial site. Additionally, it covers situations where offsite radioactive material may directly expose critical group members using the partial site release. It does not explicitly address sources or routes of exposure that result from the critical group using more than the partial release.**

## L.2 Instructions

The worksheet is split into three parts: (1) screening sources, (2) screening transport processes, and (3) exposure pathways. The method is to start with the source questions and follow the directions under each item, as necessary. The licensee should follow the path down until it screens out the item or decides to consider it in the analyses. After reaching the end of a path, the licensee should go back to where the branching occurred and continue with the questions, if applicable. For example, a site has some residual radioactivity in the soil and the licensee reviews the questions under Section L.4.2.2 (“Soil Transport: Leaching”). The questions lead to Section L.4.4, “Groundwater,” and the licensee follows that path to its conclusion. The licensee then returns to Sections L.4.2.3–L.4.2.5 to continue evaluating the source of residual radioactivity in the soil.

## L.3 Screening Sources (Yes/No)

Answer the following questions that are appropriate for each possible source of residual radioactivity.

### L.3.1 Existing/Historical Residual Radioactivity (Yes/No)

- Is residual radioactivity present in the media? (yes/no)
    - o surface soil (less than 30 centimeters (cm) (1 foot (ft))?
    - o deep soil (greater than 30 cm (1 ft))?
    - o groundwater?
    - o surface water?
    - o structures?
    - o others?

For each media type, determine if there is enough residual radioactivity to include in dose calculations? (yes/no)

- o surface soil (less than 30 cm (1 ft))? If "yes," go to Section L.4.2.
- o deep soil (greater than 30 cm (1 ft))? If "yes," go to Section L.4.3.
- o groundwater? If "yes," go to Section L.4.4.
- o surface water? If "yes," go to Section L.4.5.
- o structures? If "yes," go to Section L.4.6.
- o other? If "yes," follow the process for the most similar media.

### L.3.2 Current Operational Releases (Yes/No)

- Are there current effluents or ongoing disposals from the operating facility in the media (e.g., onsite disposals may be authorized under Title 10 of the *Code of Federal Regulations* (10 CFR) 20.2002, "Method for Obtaining Approval of Proposed Disposal Procedures")? (yes/no)
  - o gaseous or particulate release? If "yes," go to Section L.4.1.
  - o surface soil (less than 30 cm (1 ft))? If "yes," go to Section L.4.2.
  - o deep soil (greater than 30 cm (1 ft))? If "yes," go to Section L.4.3.
  - o groundwater? If "yes," go to Section L.4.4.
  - o surface water? If "yes," go to Section L.4.5.
  - o other? If "yes," follow the process for the most similar media.
- Are there ongoing or planned decommissioning activities involved with the media containing residual radioactivity? (yes/no)
  - o gaseous or particulate release? If "yes," go to Section L.4.1.
  - o surface soil (less than 30 cm (1 ft))? If "yes," go to Section L.4.2.
  - o groundwater? If "yes," go to Section L.4.4.
  - o surface water? If "yes," go to Section L.4.5.
  - o structures? If "yes," go to Section L.4.6.
  - o other? If "yes," follow the process for the most similar media.

## L.4 Screening Transport Processes (Yes/No)

Answer the following questions for the media type/source combination.

### L.4.1 Air Transport (Yes/No)

- Does the wind travel a significant portion of the year from the source to the critical group location? (yes/no)

- Is the source location near enough to the critical group location to prevent significant dilution of suspended or gaseous residual radioactivity? (yes/no)

- Do the structures, topography, and vegetation between the source and critical group locations provide only small amounts of dispersion? (yes/no)

If the answer to any one of the above questions in this section is "no," answer the following question. If all are "yes," go to Section L.5.1.

- Can air-transported residual radioactivity from this source accumulate with other source/air transport combinations that have been screened out, so that the combined effect of all sources would result in a significant source of exposure? (yes/no)

If the answer is "yes," air transport for this source and for any other sources identified by this question is not screened out. Go to Section L.5.1. If "no," air transport for the source is screened out.

### L.4.2 Surface Soil Transport (Yes/No)

Each source should go through all subsections. Screening out one subsection does not necessarily mean all subsections are screened out. To screen out the entire surface soil transport mechanism for a source, screen out Sections L.4.2.1–L.4.2.5 individually.

#### *L.4.2.1 Erosion (Yes/No)*

- Can the chemical or structural form of the residual radioactivity erode within the time frame of the analysis? (yes/no)

- Is the rainfall, runoff, or wind speed sufficient to erode source contaminants? (yes/no)

- Is the source location close enough to the critical group location for erosion to transport contaminants to the critical group location? (yes/no)

- Do the structures, topography, and vegetation between the source location and the critical group favor transport of material to the critical group location? (yes/no)

If the answer to any one of the above questions in this section is "no," answer the following question. If all are "yes," skip the next question, and then answer the last question of this section.

- Can the residual radioactivity from this source erode and accumulate with other source/erosion transport combinations that have been screened out so that the combined effect of all sources would result in a significant source of exposure? (yes/no)

If the answer is "yes," the erosion subsection for this source and any other sources identified by this question is not screened out. Answer the following question. If "no," the erosion subsection is screened out. Go to Section L.4.2.2.

- If erosion were to occur, where would the material arrive so that it can be transported to the critical group location?
    - direct overland flow? (yes/no) If "yes," go to Section L.4.5 and answer surface water questions for potential overland flow.
    - surface water body? (yes/no) If "yes," go to Section L.4.5.
    - other? (yes/no) If "yes," go to the appropriate similar transport mechanism.

If the answer to any one of these questions is "yes," the erosion subsection for this source is not screened out. Proceed as directed by the specific question. After completing that pathway, return to Section L.4.2.2. If "no," the erosion subsection is screened out. Go to Section L.4.2.2.

*L.4.2.2 Leaching (Yes/No)*

- Is the rainfall or infiltration amount sufficient for residual radioactivity to leach to a significant degree? (yes/no)
- Will the residual radioactivity leach within the time frame of the analysis? (yes/no)
- Does the geochemistry of the soil and radionuclides (e.g., distribution coefficients ($K_d$)) allow leached residual radioactivity to reach the ultimate transport mechanism within the time frame of the analysis (e.g., will the residual radioactivity be able to move through the unsaturated zone and enter into the groundwater aquifer)? (yes/no)

If the answer to any one of the above questions in this section is "no," answer the following question. If all are "yes," skip the next question, and then answer the last question of this subsection.

- Can the leached residual radioactivity from this source accumulate with other source/leach transport combinations that have been screened out so that the combined effect of all sources would result in a significant source of exposure? (yes/no)

If the answer is "yes," the leaching subsection for this source and for any other sources identified by this question is not screened out. Answer the following question. If "no," the leaching subsection is screened out. Go to Section L.4.2.3.

- If leaching were to occur, where would the material arrive so that it can be transported to the critical group location?
    - groundwater aquifer? (yes/no) If "yes," go to Section L.4.4.

- o surface water body? (yes/no) If "yes," go to Section L.4.5.

- o other? (yes/no) If "yes," go to the appropriate similar transport mechanism.

If the answer to any one of these above questions is "yes," the leaching subsection for this source is not screened out. Proceed as directed by the specific question. After completing that pathway, return to Section L.4.2.3. If the answer to each of the bullets is "no," the leaching subsection is screened out. Go to Section L.4.2.3.

*L.4.2.3 Resuspension (Yes/No)*

- Does the wind travel a significant portion of the year from the source to the critical group location? (yes/no)

- Is the source location near enough to the critical group location to prevent significant dilution of suspended or gaseous residual radioactivity? (yes/no)

- Do the structures, topography, and vegetation between the source location and the critical group favor transport of the material to the critical group location? (yes/no)

- Can enough of the residual radioactivity be resuspended to affect the dose to the critical group? (yes/no)

If the answer to any one of the above questions in this section is "no," answer the following question. If all are "yes," skip the next question and go to Section L.5.1. After completing that pathway, return to Section L.4.2.4.

- Can the resuspended residual radioactivity from this source accumulate with other source/resuspension or air-transport combinations that have been screened out so that the combined effect of all sources would result in a significant source of exposure? (yes/no)

If the answer is "yes," the resuspension subsection for this source and for any other sources identified by this question is not screened out. Go to Section L.5.1. After completing that pathway, return to Section L.4.2.4. If "no," the resuspension subsection is screened out. Go to Section L.4.2.4.

*L.4.2.4 Manual Redistribution: Excavation and Fill (Yes/No)*

- Do source area characteristics allow future excavation and reuse? (yes/no)

- Would reuse be reasonable on or near the partial site? (yes/no) A "no" on this question does not screen this subsection out.

- Would the source be able to become airborne as part of fugitive dust emissions? (yes/no) If "yes," go to Section L.4.2.3. A "no" on this question does not screen this subsection out.

If the answer to the first bullet is "no," or the second and third bullets are "no," the manual redistribution subsection is screened out. Go to Section L.4.2.5. If manual redistribution is not screened out, go to Section L.5.2. After completing that pathway, return to Section L.4.2.5.

*L.4.2.5 Direct Radiation (Yes/No)*

- Are the radionuclides significant external hazards? (yes/no)

- Is the source location close enough to the critical group location to prevent a significant reduction in the dose rate? (yes/no)

- Do the structures, topography, and vegetation between the source and critical group locations provide inadequate shielding to minimize the external exposure? (yes/no)

If the answer to any one of the above questions in this section is "no," the direct radiation subsection is screened out. If all are "yes," go to Section L.5.2.

## L.4.3 Deep Soil Transport (Yes/No)

Each source should go through both subsections below. Screening out one subsection does not necessarily mean that both subsections are screened out. To screen out the entire deep soil transport mechanism for a source, screen out both Section L.4.3.1 and Section L.4.3.2 individually.

*L.4.3.1 Leaching (Yes/No)*

- Is the rainfall or infiltration amount sufficient for residual radioactivity to leach to a significant degree? (yes/no)

- Will the residual radioactivity leach within the time frame of the analysis? (yes/no)

- Does the geochemistry of the soil and radionuclides (e.g., $K_d$) allow leached residual radioactivity to reach the ultimate transport mechanism within the time frame of the analysis (e.g., will the residual radioactivity be able to move through the unsaturated zone and enter the groundwater aquifer)? (yes/no)

If the answer to any one of the above questions in this section is "no," answer the following question. If all are "yes," skip the next question, and then answer the last question of this subsection.

- Can leached residual radioactivity from this source accumulate with other source/leach transport combinations that have been screened out so that the combined effect of all sources would result in a significant source of exposure? (yes/no)

If the answer is "yes," the leaching subsection for this source and for any other sources identified by this question is not screened out. Answer the following question. If "no," the leaching subsection is screened out. Go to Section L.4.3.2.

- If leaching were to occur, where would the material arrive so that it can be transported to the critical group location?

    - groundwater aquifer? (yes/no) If "yes," go to Section L.4.4.

    - surface water body? (yes/no) If "yes," go to Section L.4.5.

- other? (yes/no) If "yes," go to the appropriate similar transport mechanism.

If the answer to any one of these is "yes," the leaching subsection for this source is not screened out. Proceed as directed by the specific question. After completing that pathway, return to Section L.4.3.2. If the answers to all of the bullets are "no," the leaching subsection is screened out. Go to Section L.4.3.2.

*L.4.3.2 Manual Redistribution: Excavation and Fill (Yes/No)*

- Do source area characteristics allow future excavation and reuse? (yes/no)
- Would reuse be reasonable on or near the partial site? (yes/no) A "no" on this question does not screen this subsection out.
- Could the source become airborne as part of fugitive dust emissions? (yes/no) If "yes," go to L.4.2.3. A "no" on this question does not screen this subsection out.

If the answer to the first bullet is "no," or all bullets are "no," the manual redistribution subsection is screened out. If manual redistribution is not screened out, go to Section L.5.2.

**L.4.4 Groundwater Transport (Yes/No)**

- Is saturated groundwater in hydraulic connection with the radioactive source? (yes/no)
- Does the groundwater (including unconfined or confined aquifers, as necessary) flow from the source to the location of the critical group? (yes/no)
- Is the aquifer fit for use? (yes/no)
- Is the aquifer potable? (yes/no)
- Is groundwater used for irrigation? (yes/no)
- Can the residual radioactivity enter the groundwater aquifer in significant amounts (e.g., is the aquifer not protected from all potential migrating contaminants by low-permeability geologic strata (e.g., clay layer))? (yes/no)
- Is the yield rate of the aquifer sufficient? (yes/no)
  - household and drinking water? (yes/no)
  - irrigation? (yes/no)
- Is the distance traveled from the source to the critical group location close enough to prevent significant dilution and sorption of migrating radionuclides? (yes/no)

If the answer to any of the above questions (not related to irrigation) in this section is "no," the drinking water/groundwater transport mechanism is screened out. If the questions related to irrigation are no, the irrigation/groundwater transport mechanism is screened out. If all are "yes," go to Section L.5.3.

### L.4.5 Surface Water Transport (Yes/No)

- Does surface water flow from the source of residual radioactivity (or from zones of mobilized radionuclides) to the critical group location? (yes/no)

- Does the volume of surface water allow transport of significant concentrations of either dissolved or suspended radioactive solids? (yes/no)

If the answer to either of the above questions in this section is "no," the surface water transport mechanism is screened out. If both are "yes," answer the following question.

- Is significant sediment buildup possible at the critical group location? (yes/no)

If the answer is "yes," go to Section L.5.5. After completing that pathway, return to Section L.5.4. If the answer is "no," go to Section L.5.4.

### L.4.6 Structures (Yes/No)

#### *L.4.6.1 Direct Radiation (Yes/No)*

- Are the radionuclides significant external hazards? (yes/no)

- Is the source location close enough to the critical group location so that a significant dose rate cannot be avoided? (yes/no)

- Do the structures, topography, and vegetation between the source and critical group locations provide inadequate shielding to minimize the external exposure? (yes/no)

If the answer to any one of the above questions in this section is "no," the direct radiation subsection is screened out. Go to Section L.4.6.2. If all are "yes," go to Section L.5.2. After completing that pathway, return to Section L.4.6.2.

#### *L.4.6.2 Leaching (Yes/No)*

- Is the rainfall or infiltration amount sufficient for residual radioactivity to leach to a significant degree? (yes/no)

- Will the residual radioactivity leach from the structure within the time frame of the analysis? (yes/no)

- Does the geochemistry of the soil and radionuclides (e.g., $K_d$) allow leached residual radioactivity to reach the ultimate transport mechanism within the time frame of the analysis (e.g., will the residual radioactivity be able to move through the unsaturated zone and enter the groundwater aquifer)? (yes/no)

If the answer to any one of the above questions in this section is "no," answer the following question. If all are "yes," skip the next question, and then answer the last question of this subsection.

- Can the leached residual radioactivity from this source accumulate with other source/leach transport combinations that have been screened out so that the combined effect of all sources would result in a significant source of exposure? (yes/no)

If the answer is "yes," the leaching subsection for this source and for any other sources identified by this question is not screened out. Answer the next question. If "no," the leaching subsection is screened out.

- If leaching were to occur, where would the material arrive so that it can be transported to the critical group location?
    - groundwater aquifer? (yes/no) If "yes," go to Section L.4.4.
    - surface water body? (yes/no) If "yes," go to Section L.4.5.
    - other? (yes/no) If "yes," go to the appropriate similar transport mechanism.

If the answer to any one of these questions is "yes," the leaching subsection for this source is not screened out. Proceed as directed by the specific question. If the answers to all these dash bullets are "no," the leaching subsection is screened out.

## L.5 Exposure Pathways

"No" on the black bullets will not eliminate the entire section.

### L.5.1 Air Pathways (Yes/No)

- Based on the critical group habits and activities, are the following viable? (yes/no)
    - inhalation? (yes/no)
    - air submersion external dose? (yes/no)
- Is significant deposition viable? (yes/no) If "yes," go to Section L.5.2 and consider the potential soil pathways at the deposition area.

### L.5.2 TSoil Pathways (Yes/No)

- Is external exposure viable? (yes/no)
- Is exposure through ingestion viable? (yes/no)
    - direct soil ingestion? (yes/no)
    - garden or crops? (yes/no)
    - leafy vegetables? (yes/no)
    - nonleafy vegetables? (yes/no)
    - fruits? (yes/no)

- grain? (yes/no)
- animal husbandry? (yes/no)
- meat? (yes/no)
- milk? (yes/no)
- eggs? (yes/no)

- Is exposure through inhalation viable? (yes/no)
  - indoors? (yes/no)
  - outdoors? (yes/no)

### L.5.3 Groundwater Pathways (Yes/No)

- Is exposure via drinking water viable? (yes/no)
- Is exposure via irrigation viable? (yes/no)
  - garden or crops? (yes/no)
  - animal husbandry? (yes/no)
  - fish farming? (yes/no)

<u>If irrigation is viable, go to Section L.5.2. Consider the soil pathways appropriate for the soil affected by the irrigation</u>.

- Is water used for purposes other than household uses (including drinking water) or irrigation (e.g., evaporative coolers, dust suppression)? (yes/no)

<u>If "yes," go to Section L.5.2. Consider the soil pathways appropriate for the impacts of the activity</u>.

### L.5.4 Surface Water Pathways (Yes/No)

- Is internal exposure viable? (yes/no)
  - fish? (yes/no)
  - drinking water? (yes/no)
  - inadvertent intakes? (yes/no)
  - via irrigation? (yes/no)
  - garden or crops? (yes/no)
  - animal husbandry? (yes/no)

If irrigation is viable, go to Section L.5.2. Consider the soil pathways appropriate for the soil affected by the irrigation.

- Is water used for purposes other than household uses (including drinking water) or irrigation (e.g., evaporative coolers, dust suppression)? (yes/no)

If "yes," go to Section L.5.2. Consider the soil pathways appropriate for the impacts of the activity.

- Are recreational activities viable? (yes/no)

If recreational activities are viable, go to Section L.5.2. Consider the exposure pathways appropriate for recreational activities in the water (e.g., incidental ingestion during swimming).

### L.5.5 Sediments (Yes/No)

- Are recreational activities viable? (yes/no)

If recreational activities are viable, go to Section L.5.2. Consider the exposure pathways appropriate for recreational activities on the shore, or involving sediments (e.g., incidental ingestion from making sand castles).

- Is use of sediments for land-based activities viable (e.g., fill or crops)?

If use of sediments is viable, go to Section L.5.2. Consider the soil pathways appropriate for the impacts of the activity.

### L.5.6 Documentation

The information from the worksheet should be summarized in tables. The tables should include (1) the source, (2) whether it is included or excluded, (3) the FEPs screened, (4) the screening argument, and (5) the reference for the screening argument. For example, one format shown below uses the example in Section 3.1 of Appendix K as a basis. The level of detail is only needed for the question being used to screen out the source, transport mechanism, or pathway. Common pathways using the same or similar screening arguments can be grouped (e.g., fourth row of the example below).

**Table L.1 Example of Summary Format**

| Source | Status | Screening Pathway[a] | Screening Argument | Reference |
|---|---|---|---|---|
| Groundwater (GW) Plume–Area 4-10 | Incl | GW (3.1)–GW (4.4)–GW (5.3)–Soil (5.2) | N/A | N/A |
| GW Plume–Area 4-12 | Excl | GW (3.1)–GW (4.4/YIELD) | Yield of Pico Aquifer <30 L/day. | DP Chapter 3.7.3 |
| Low-Level Waste (LLW) Storage Area | Incl | Other (3.2)–Soil (4.6)–Soil (5.2/direct) | N/A | N/A |
| | Excl | Other (3.2)–Soil (4.1–4.2) | No significant erosion or leaching of LLW area within 1,000 years. | DP Chapter 4.1.5 |

Note:
a. Numbers in this column indicate the appropriate sections in Appendix L.

# APPENDIX M

# PROCESS FOR DEVELOPING EXPOSURE SCENARIOS USING SITE-SPECIFIC INFORMATION

## M.1 Introduction

This appendix discusses modification of the two default or screening-level exposure scenarios, the residential farmer and the building occupancy exposure scenarios, using site-specific information. The residential farmer exposure scenario is applied to sites with land and water residual radioactivity, while the building occupancy exposure scenario is applied to sites with contaminated structures. A generic critical group, with associated default parameter representing the average member of each group, is associated with each exposure scenario. The default pathways, models, and parameter values for the critical group combine to form exposure scenarios.

The default exposure scenarios may not always be appropriate or may lead to overly conservative results. For example, there is significant variability among decommissioning sites with respect to physical features and characteristics of residual radioactivity remaining on site. The original purpose of the site, historical development, and the resulting processes that generated the site's residual radioactivity vary widely. Residual radioactivity may occur in buildings, process equipment and other site structures, soils (surface and subsurface), ponds, lagoons, surface water, and groundwater. Sites are in urban and suburban; residential, commercial and industrial, rural, and agricultural areas; and many are located on or directly adjacent to rivers, lakes, oceans, estuaries, wetlands, flood plains, or wildlife areas. The physical and chemical form of residual radioactivity is highly variable. Residual radioactivity may be associated with slag, soils, sediments, sludge, debris, dust or sand piles, packaged waste (e.g., drums, crates), and dispersed in liquid media. These factors should be considered in developing reasonably foreseeable exposure scenarios, as well as those that are considered less likely but plausible.

This appendix steps through the process of developing alternative exposure scenarios and for eliminating various default pathways from the residential farmer exposure scenario and describes the information needed to support these modifications. Contact with state and local agencies is recommended early in the process to obtain information on state and local laws and regulations, and sources of information to support development of alternative scenarios and elimination of pathways. The process for developing alternative exposure scenarios complements the Decision-making Framework and is meant to be used in conjunction with the methodology discussed in Chapter 1 of this volume and in the guidance on exposure scenarios, exposure pathways, and critical groups discussed in Appendix I, Section I.3.

While development of site-specific exposure scenarios is always an option, if screening assumptions and values are met, there may be no need to collect additional data to develop alternative exposure scenarios. If the dose criteria of 10 CFR 20.1402 are met through use of the screening values and the residual radioactivity has been reduced to levels that are ALARA, the site would be considered a candidate for unrestricted use. However, if screening level analyses do not support compliance with the release standard, then one option is to use site-specific information to modify the resident farmer exposure scenario, including use of site-specific information to support elimination of pathways that are inappropriate for the site in question. Sensitivity analysis would be helpful in determining what pathways and parameters have a significant effect on the dose. While other options are available to meet release criteria (e.g., additional source removal), the focus of this appendix is on the development of alternative exposure scenarios.

## M.2 Introducing Site-Specific Information to Modify or Develop Alternate Exposure Scenarios

Site-specific information can be divided into two broad categories: cultural information and physical information. Physical information includes such characteristics as the location, climate, topography, geology, soil types, and water availably of the site. Cultural information is basically how the land is used by the human population. Physical properties of land are essentially unchanging, while cultural properties are constantly changing. In reality, physical properties change (sometimes as a result of cultural activities), but the change is slow compared to the cultural use of the land.

Since the initial dose assessment for this process used the resident farmer exposure scenario with NRC–approved default pathways and parameters, the introduction of either cultural or physical information about a decontamination and decommissioning site is likely to reduce the TEDE.

While the two screening scenarios are expected to be conservative, it should be noted that they might be less conservative than expected. Therefore, care should be taken when eliminating exposure pathways and exposure scenarios should be consistent with site conditions.

### M.2.1 Cultural Information

For developing alternative exposure scenarios, the most important element of cultural information about any site is the future land use over the period residual radioactivity at a site will persist at risk-significant levels. The future is assessed based on the past and the present. Experience has shown that, while this is an inexact science, the near future can be estimated with some degree of accuracy. Although the near future depends on the location, the culture, and what is being estimated, for the purposes of this section the near future is considered 100 years; future land use over longer periods of time are uncertain and may not need to be considered. The key to the assessment of future land use is the current and past use of the land at both the site and in the region. If future land use is reasonably expected to be either urban or industrial, the resident farmer scenario may potentially be omitted from evaluation as a compliance scenario, although it may be evaluated to provide additional information for the decision-making process (i.e., as a less likely but plausible exposure scenario). The licensee will need to address both the dose from reasonably foreseeable land use (for the compliance calculation) and less likely but plausible land uses (to risk-inform the decision on whether release criteria are met). With adequate justification, pathways may be eliminated from detailed consideration in dose assessments. However, care should be taken to ensure that spatial and temporal variability is considered when justifying elimination of pathways (e.g., if poor water quality is used to justify elimination of the groundwater pathway, then the analyst should consider whether poor water quality conditions are expected to persist into the future).

#### *M.2.1.1 Current Land Use*

The determination of current land use is the initial step in the process of estimating future land use. Land use should be determined not only for the site but also for the land within an 80-km (50-mile) radius surrounding the site. This assessment of land use does not need to be complicated or detailed; it should be simple, dividing the land into only three categories: urban, rural, or industrial.

Current land use can be determined through one or more of the following information sources:

- site description
- land cover maps
- topographic maps
- planning agencies
- zoning laws and maps
- land use plans
- aerial photographs or satellite imagery
- demographic data
- cultural and natural resource data
- site visits

Most of the United States has codified land use/zoning, and many administrative areas have developed land use master plans. For this reason, the primary source of information on current land use should be the planning agencies of the State, county, and municipality in which the site resides.

A large amount of land use information is available on the Internet at websites maintained by government agencies or universities. Land use data is often collected, stored, and analyzed using a GIS. An Internet search that includes the name of the government entity and the phrase "GIS data" or "land use data" will usually find the website where GIS data, including land use data, can be downloaded or ordered. For locations where State or local land use data are unavailable, data collected by one or more Federal agencies, such as the USGS, U.S. Department of Agriculture (USDA), or the U.S. Census Bureau, can usually be found. In particular, the Multi-Resolution Land Characteristics Consortium maintains nationwide land cover data, from which land use can be inferred, via the National Land Cover Database. This program is a joint effort of numerous Federal agencies, including USGS, USDA, EPA, Bureau of Land Management, National Oceanic and Atmospheric Administration, National Park Service, U.S. Forest Service, U.S. Fish and Wildlife Service, National Aeronautics and Space Administration, and the U.S. Army Corps of Engineers. The National GAP Analysis Program is also a potential source of land cover data for much of the United States.

Assumptions and predictions on future land uses are important considerations in the development of exposure scenario definitions and descriptions for analysis. If the site currently exists in a highly populated urban area, a residential farmer exposure scenario is very unlikely. Exposure scenarios for certain sites may exclude exposures via agricultural pathways if agricultural land uses are clearly incompatible with existing and anticipated future conditions at the sites. Exposures via ingestion of contaminated groundwater may be discounted if the affected groundwater is of such poor quality as to preclude human consumption.

### *M.2.1.2 Use of Ponds as Fisheries*

In addition to physical limitations on the likelihood of a farmer using a pond as a fishery, the licensee should use local cultural information to determine if local residents currently engage in this practice. This question might be answered by the USDA county extension agent nearest to the decommissioning site.

In addition to exposure of the average member of the critical group through use of a pond as a fishery, contaminated groundwater could also discharge to surface water as conceptualized in RESRAD-ONSITE more generally (see Appendix I). In contrast to the DandD conceptual model, RESRAD-ONSITE allows dilution in surface water, based on the ratio of the watershed area to the area of the contaminated zone.[1] If the licensee eliminates the use of the pond as a fishery, based on site-specific information, it should consider if the surface water pathway is otherwise viable (e.g., discharge of groundwater to a site stream used for fishing).

### *M.2.1.3 Future Land Use*

Licensees should consider specific local conditions when deciding how far into the future it can estimate land use. The general range for estimation is within 100 years. In areas where rapid change has occurred in the past, this cutoff might be considerably less than 100 years, whereas in other areas, such as the heart of New York City, it may be reasonable to argue that urban conditions should prevail for more than 100 years.

The first step in estimating future land use is to determine the current land use at the site. The licensee should also learn the past use of the land, because it is the combination of past and present uses that should indicate what changes have occurred and the rate of those changes. This information should be used in a documented process that a reviewer would be able to follow. This documentation should include the types and sources of material used and how the licensee determined the final projected use.

Land use and changes in land use within the 80-km (50-mile) radius of the site are included in this process. For example, a site that is currently located in a rural area within 16–32 km (10–20 miles) of a growing metropolitan area would likely be in the suburbs of the metropolitan area within a decade or two, depending on population growth.

The 80-km (50-mile) radius is only a suggestion for determining the size of the area to consider. There may be valid reasons for increasing or decreasing the area of consideration, depending on local conditions and the length of time the residual radioactivity presents a potential risk. Other factors that may influence this decision are critical pathways and the estimated distribution of residual radioactivity.

### *M.2.1.4 Sources of Information for Determining Future Land Use*

EPA's Office of Solid Waste and Emergency Response (OSWER) publishes useful guidance related to land use assessments including OSWER Directive No. 9355.7-04, "Land Use in the CERCLA Remedy Selection Process," dated May 25, 1995, and OSWER 9355.7-06P, "Reuse

[1] RESRAD-OFFSITE 4.0 contains a more sophisticated surface water model that considers contamination of surface water via (i) sediment erosion and runoff, (ii) groundwater discharge, and (iii) airborne deposition. The code considers the water balance, sediment balance, and radionuclide balance of the surface water body in order to compute the aqueous concentration of radionuclides in the surface water body.

Assessments: A Tool to Implement the Superfund Land Use Directive," dated June 4, 2001. Published in 2010, OSWER 9355.7-19 reaffirms OSWER 9355.7-04 indicating that the 1995 directive continues to provide useful guidance on consideration of reasonably anticipated future land use in the Superfund remedy selection process. OSWER 9335.7-04 and -19 are intended to facilitate future remedial decisions at Superfund sites by outlining a public process and sources of information which should be considered in developing reasonable assumptions regarding future land use. Land use is integral to determining the efficacy of the remedy with respect to long-term effectiveness, protectiveness and compliance with all applicable or relevant and appropriate requirements.

Because so much of the information used to describe current land uses and to determine possible future land uses is geographic in nature, the sources include GIS providers at both the national and State levels. State GIS organizations should be able to direct the licensee to local sources for much of this information and, in many cases, may have links to that information directly from their data sites.

Land use information types include the following:

- zoning laws and maps
- state and community comprehensive master plan
- demographics
- historical population growth patterns
- current site location relative to other land uses
- federal/state land designations for surrounding lands
- threatened and endangered species
- natural resource inventory information
- floodplain/wetlands designations
- local/regional geologic information
- wellhead protection areas/aquifer recharge areas
- state comprehensive groundwater protection program
- historical aerial photography
- environmental justice issues

Federal agencies managing data that may be relevant to land use include the following:

- United States Geological Survey

- Multi Resolution Land Characteristics Consortium (MRLC) National Land Cover Dataset (NLCD)
- Water Resources Division (WRD) National Spatial Data Infrastructure (NSDI) Node
- EarthExplorer
- GAP Analysis Project

- United States Department of Agriculture
  - National Resources Conservation Service (NRCS)
  - National Agricultural Statistics Service (NASS)
  - Economic Research Service (ERS)
  - Forest Service (USFS)
- EPA
- National Oceanic and Atmospheric Administration
- United States Department of Interior
  - Fish and Wildlife Service National Wetlands Inventory (NWI)
  - Bureau of Land Management (BLM)
- Census Bureau
  - TIGER data
- U.S. Department of Transportation

#### *M.2.1.5 Urban Gardens*

The subsistence farm associated with the resident farmer is unlikely to exist in an urban situation, but gardens are very likely in urban and suburban settings. The "Victory Gardens" of World War II demonstrate this possibility. Exceptions would be places like the concrete and steel core of large cities like New York, where gardens using locally obtained soil and water would be highly unlikely.

**Documentation to Be Submitted to the NRC**

**Current Land Use** should be documented by maps, descriptions, or other information

**Estimates of Future Land Use** should be supported by the documented process described in Sections M.2.1.3 and M.2.1.4.

State and local agencies should be contacted to get information on local land use and restrictions.

## M.2.2 Physical Information about the Site

Physical information about the site includes climate, topography, vegetation, and, most importantly, water. Since water is a key factor in many of the pathways, its availability and proximity are very important.

### *M.2.2.1 Groundwater and Surface Water*

Groundwater is present at some depth at almost every site. If groundwater is only found at great depths, surface water may be ephemeral and may exist only in response to rainfall or snowmelt. Surface water for the resident farmer may be a fish pond that is connected to the groundwater.

Several key questions about groundwater should be answered using site-specific information. The most important question is on the availability of water. Subsequent questions concern its quality and suitability for use.

### *M.2.2.2 Is Groundwater Available?*

The first question that should be answered is "Is groundwater available as a resource for the exposure scenario resident?" More specific questions include the following:

(1) Is it shallow enough and does it have sufficient yield to be reasonably pumped by the resident to irrigate a small farm and provide domestic drinking water?

(2) Is it shallow enough to intercept and connect to a fish pond, and does it have sufficient yield to sustain the pond?

For the first question, the resident would need to drill a well into a permanent aquifer that has water sufficient for his needs and then be able pump that water into his house and onto his crops. Under the assumption that the well drilling and pumping technology available to the resident is similar to what exists today, it would not be unreasonable for the farmer to drill a well and pump from a depth of 120 m (400 ft), but this depth should be considered somewhat subjective. Specific local conditions should be considered when deciding how deep an aquifer a subsistence farmer would be able to use. A commercial farmer would be likely to drill much deeper than a subsistence farmer would.

Local trends in groundwater decline should be considered. In areas where groundwater is being withdrawn at an unsustainable rate, water levels may be dropping. If it can be reasonably assumed that this trend may continue, the licensee should consider it when assessing the availability of groundwater for the resident farmer.

If groundwater is not available at a reasonable depth for drinking water or irrigation, it may also not be available for a pond. Under these circumstances, the resident farmer exposure scenario can be devolved to exclude all three of the major pathways based on groundwater usage: irrigation, drinking water, and aquatic (pond). If groundwater is unavailable, it is also reasonable to exclude the use of surface water, since the aquatic exposure scenario considers the concentration of radionuclides in the surface water to be related to the concentration in the groundwater aquifer (NUREG/CR-5512, Volume 1, "Residual Radioactive Contamination from Decommissioning: Technical Basis for Translating Contamination Levels to Annual Total Effective Dose Equivalent," 1992 (Kennedy, 1992).

Even if groundwater were available at a reasonable depth, the yield may be insufficient for all uses. Under these circumstances, the resident farmer exposure scenario can be modified to be more realistic considering the yield (e.g., pathways can be eliminated, or parameters adjusted to account for the lower yield). State and local agencies should be contacted to get input on groundwater use restrictions, groundwater protection regulations, and for any objections to elimination of groundwater related pathways.

**Documentation to Be Submitted to the NRC**

**Groundwater Unavailable**: A USGS or independent consultant report showing that either groundwater does not exist, or that it is too deep (e.g., more than 120 m (400 ft)) to reasonably be used by a subsistence farmer.

State and local agencies should be contacted to get input on groundwater use restrictions, groundwater protection regulations, and support for elimination of groundwater related pathways.

If groundwater is available for drinking or irrigation, it may not be available for a fish pond. It would not be reasonable to expect that the farmer would continually pump water into a pond to maintain it as a fishery. The groundwater would have to be shallow enough that a sufficient pond level would be maintained through its connection to the pond. This would mean the groundwater would have to be no deeper than about 5 m (15 feet). Information about local topography and specific conditions at each site could be used to adjust this number up and down. If groundwater is not available for the pond, the aquatic pathway should be removed from the resident farmer exposure scenario.

**Documentation to Be Submitted to the NRC**

**Groundwater Unavailable**: A USGS or independent consultant report showing that either groundwater does not exist, or that it is too deep (e.g., more than 5 m (15 ft)) to connect to a surface water pond.

State and local agencies should be contacted to get information on local land use practices, and to support elimination of the surface water pathway from consideration.

*M.2.2.3 Is Groundwater Quality Suitable for Aquatic Life?*

The quality of surface water is critical to support aquatic life and is affected by (1) the chemical and physical conditions that exist in the pond, (2) runoff from exposed soil, and (3) condensation

or entrapment of contaminants from the air (e.g., pollutants, acid rain). EPA provides recommended surface water standards to support aquatic life, including minimum dissolved oxygen values and maximum contaminant levels (MCLs).

The concentration of dissolved oxygen in surface water is affected by the biochemical oxygen demand of the ecosystem. Sedimentation of suspended solids can cause a buildup of organic matter in sediments. These materials undergo metabolic degradation by aerobic soil microorganisms with the concomitant depletion of dissolved oxygen. Other contaminants, such as dissolved ammonia, can contribute to oxygen depletion by nitrification. Ammonia is toxic to fish and other aquatic animals. The presence of coliform bacteria is sometimes indicative of other, more virulent pathogens in surface water and should be considered when fish or other aquatic animals are produced for human consumption. If the quality of the groundwater (and hence the pond) lies outside of the acceptable standards for aquatic life, the licensee can remove the aquatic pathway from the resident farmer exposure scenario.

**Documentation to Be Submitted to the NRC**

**Groundwater Unsuitable for Aquatic Life**: A USGS or independent consultant report showing that groundwater quality is poorer than the standards listed for this use.

State and local agencies should be contacted to get supporting information on elimination of the aquatic pathway based on groundwater quality.

### *M.2.2.4 Is Groundwater Quality Suitable for Agriculture?*

The quality of groundwater for agricultural uses varies depending on the type of agribusiness or agricultural enterprises conducted at the site. For example, groundwater with infiltrated fertilizers and herbicides can be very beneficial to crop land through irrigation but can have an adverse effect on the health and productivity of livestock and poultry. Based on extensive USDA studies, recommended limits for chemicals in drinking water for livestock and poultry have been published and are available in an Internet search.

In addition to acute and chronic toxicity from chemicals, high concentrations of dissolved solids in drinking water can lead to various degrees of mineral toxicity in animals. Most minerals and dissolved solids found in water provide nutritional benefits when present within limited concentration ranges (e.g., selenium). At high concentrations, however, common minerals can lead to acute or chronic changes that affect the quality of animal products and overall productivity.

The licensee should consider salinity, or total dissolved solids, when evaluating groundwater for animal consumption. Although 10,000 mg/L is acceptable under some conditions, the health, and ultimately the productivity, of animals are affected to various degrees by the salinity. Table M.1 provides a breakdown of conditions that have been observed and documented in livestock and poultry for various concentrations of dissolved solids in drinking water.

**Table M.1 Effects of Salinity of Drinking Water on Livestock**

| Salinity Level Limits for Drinking Water | Conditions |
|---|---|
| Less than 1,000 mg/L | Excellent for all classes of livestock and poultry |
| 1,000–3,000 mg/L | Satisfactory for all classes of livestock. May cause temporary and mild diarrhea in livestock and poultry not accustomed to such levels but should not affect their health or performance. |
| 3,000–5,000 mg/L | Satisfactory for livestock, although they might very possibly cause mild diarrhea or be refused by animals not accustomed to such levels. Increased morbidity and decreased growth in poultry. |
| 5,000–7,000 mg/L | Marginal quality for livestock. Not suitable for poultry and pregnant and lactating animals. |
| 7,000–10,000 mg/L | Not suitable for pigs, and considerable risk for pregnant and lactating animals. In general, their use should be avoided. |
| Above 10,000 mg/L | Unacceptable |

If the quality of the groundwater is less than what is considered acceptable for irrigation, the licensee should remove the irrigation pathway from the resident farmer exposure scenario.

If the quality of the groundwater is less than what is considered acceptable as a drinking source for farm animals, the licensee should remove that pathway from the resident farmer exposure scenario.

**Documentation to Be Submitted to the NRC**

**Groundwater Unsuitable for Agriculture**: A USGS or independent consultant report showing that groundwater quality is poorer than the standards listed for this use.

State and local agencies should be contacted to get supporting information on elimination of use of groundwater for irrigation or livestock consumption based on quality.

### *M.2.2.5 Is Groundwater Suitable for Drinking Water?*

The licensee can address this question by comparing the quality of the groundwater with EPA drinking water standards. Regulations for public water systems in the United State are in Title 40 of the *Code of Federal Regulations* (40 CFR) Part 141, “National Primary Drinking Water Regulations.” Primary drinking water standards specify approval limits for microorganisms, including bacteria and viruses, specific inorganic and organic chemicals,

radionuclides, and turbidity, while secondary standards identified in 40 CFR Part 143, "National Secondary Drinking Water Regulations," recommend limits on benign contaminants and define physical characteristics that address the aesthetics of drinking water (e.g., color and odor).

Drinking water standards are available for (1) inorganic chemicals, (2) organic chemicals, (3) radionuclides, and (4) microorganisms. Although turbidity is a measured physical parameter, it is included with microorganisms, because turbid water is generally associated with microorganisms or provides a medium for microbial growth. Although the secondary standards are not regulated, they serve as a guide for water quality and may, in some instances, be regulated at the State or local level.

**Documentation to Be Submitted to the NRC**

**Groundwater Not Potable**: A USGS or independent consultant report that shows that groundwater quality is poorer than the standards listed for this use.

EPA regulations should be used to support elimination of the groundwater pathway based on groundwater quality. State and local agencies should be contacted to get information to support elimination of the groundwater pathway.

## M.2.3 Topography and Soil

### *M.2.3.1 Is Soil Suitable for Agriculture?*

Soil performs several functions related to plant growth. It forms a media in which roots penetrate, thereby providing a source of stability and nourishment. Nourishment can be provided by the nutrients available in the soil, by fertilizers, or by soil amendments.

With suitable fertilizers or soil amendments, plants can readily be grown in "soil free" materials, such as mineral sand, gravel, perlite, pumice, crushed bricks, or glass wool. Consequently, the absence of soil in the traditional sense at a site does not eliminate plant ingestion as a pathway. Because soilless gardening requires more management than traditional gardening methods, it is more likely to be used for growing vegetables and herbs than for the production of commodity items such as grains or livestock fodder (R.E. Nicolls, "Beginning Hydroponics Soilless Gardening," 1997 (Nicholls, 1997)). Agriculture could be excluded from an exposure scenario if the site is an outcropping of bedrock without appreciable soil or debris that could serve to anchor plants.

Areas consisting of made land, where there is abundant debris and cobbles with little or no soil, would not lend themselves to mechanized agriculture in short-term exposure scenarios. In the absence of mechanized agriculture, commodity food items and fodder are not likely crops. However, it would be difficult to exclude vegetable gardens from exposure scenarios at such sites. In addition, it would be difficult to justify exclusion of livestock forage from exposure scenarios for such sites.

Agriculture pathways could be eliminated in short-term exposure scenarios if the soil is outright toxic or inhospitable to plants. As examples, (1) no agriculture is apt to occur on the bed of a dry salt lake, and (2) crops are not apt to be grown in made land that contains such a high

percentage of concrete materials that extraordinary efforts would be required to maintain the soil pH in a range that is tolerated by plants.

If it can be documented that the soil at this site would not support the resident farmer's agricultural efforts, the licensee could eliminate or modify this pathway.

**Documentation to Be Submitted to the NRC**

Soil Unsuitable for Agriculture: A Natural Resources Conservation Service (Soil Conservation Service) or independent consultant report that shows the quality of soil is poorer than the standards listed for this use.

State and local agencies should be contacted to get information to support elimination of agricultural pathways based on soil suitability.

*M.2.3.2 Is Topography Suitable for Agriculture?*

In the past few hundred years, dikes have been used to convert submerged land into productive farmlands. Today, explosives and earth-moving equipment can easily change features of the landscape, making them suitable for agricultural or residential use. Consequently, locality or accessibility may form a basis for eliminating certain agricultural pathways from exposure scenarios in the next century but not for a period of 1,000 years.

Ignoring the fact that topography may change with time as a result of civil engineering projects, there are probable limits to the types of terrain where mechanized agriculture can be used. Tractors are likely to always be unstable on slopes, so there is probably a practical limit on slopes that can be put under mechanized agriculture. While there is no predictable maximum safe slope that tractors may traverse without the danger of rollover, operating a tractor on a 30-degree (2 to 1) slope is so hazardous that the average member of the critical group is not likely to attempt it.

In the absence of mechanized agriculture, persons are more likely to practice gardening than to grow commodity food items. In fact, gardening is commonly practiced on hillsides using contour rows, terraces, or raised beds to minimize erosion. They may also be used to allow livestock to forage.

If the topography at the site is too steep or too erratic to support the type of farming expected within the resident farmer exposure scenario, the licensee could justify modification of the agricultural pathway in accordance with this finding. There may also be aspects of the topography that would limit farming or other specific activities at the site.

**Documentation to Be Submitted to the NRC**

**Topography Unsuitable for Agriculture**: A USGS or similar topographic map, hand-drawn map, or description with enough detail to illustrate the topography that limits farming at this site.

State and local agencies should be consulted regarding local land use practices and to obtain information to support modification of the agriculture pathway based on topography.

## M.3 Summary

The process presented in this appendix is compatible with the Decommissioning Framework discussed in Chapter 1. It discusses use of site-specific information to develop alternative exposure scenarios by eliminating pathways from the default resident farmer exposure scenario. Once the TEDE to an average member of the critical group drops below 0.25 mSv/y (25 mrem/y), the process is complete, and the licensee may proceed to license termination. Following the initial dose assessment and each of the iterative dose assessments, sensitivity analyses help the licensee focus on the introduction of evidence that can rule out those pathways that are responsible for the high dose.

The licensee uses physical and cultural information to answer a series of questions about the site. The future use of the land may be important in deciding what assumptions the licensee can make about the starting exposure scenario. Information on current land use, past land use, and a history of land use changes can determine the probable future use of the land. If the future land use can reasonably be predicted to be either urban or industrial, the resident farmer exposure scenario can be bypassed, allowing the licensee to concentrate on these other two land uses.

The residential farmer exposure scenario is meant to be applied to sites with land and water residual radioactivity, and the building occupancy exposure scenario is to be applied to sites with contaminated structures. If a resident farmer exposure scenario is assumed, the most important aspect of the physical nature of the site is the nature and availability of water. The answers to each of four critical questions about water at the site can determine if major pathways can be removed from the exposure scenario. If groundwater is not available, the licensee can remove all of the pathways that rely on groundwater as a key component (irrigation, aquatic, and drinking). If groundwater is not suitable for aquatic life, the aquatic pathway can be removed. If groundwater is not suitable for agriculture, irrigation and drinking water pathways can be removed. If the water is not potable, the drinking water pathway can be removed. Detailed discussion in this appendix helps the licensee answer these questions, understand the standards that it would have to meet to rule out this pathway, and identify the documentation that it would have to present to the NRC.

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# APPENDIX N

# ALARA ANALYSES

## N.1 Introduction

To terminate a license, a licensee must demonstrate that it has met the dose criteria in Subpart E and the requirements for reducing exposures ALARA. This section describes methods acceptable to the NRC staff for determining when it is reasonable to further reduce the (future) exposures of members of the public below the dose criteria. This section does not apply to, nor replace guidance for, operational ALARA programs. This guidance does involve the same principle as the operational ALARA guidance, as described in NRC Regulatory Guide 8.8, Revision 3, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low as Is Reasonably Achievable, issued June 1978:

> "Reasonably achievable" is judged by considering the state of technology and the economics of improvements in relation to all the benefits from these improvements. (However, a comprehensive consideration of risks and benefits will include risks from non-radiological hazards. An action taken to reduce radiation risks should not result in a significantly larger risk from other hazards.)

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

In light of the conservatism in the building surface and surface soil generic screening levels developed by the NRC, the staff presumes, absent information to the contrary, that licensees who remediate building surfaces or soil to the generic screening levels (see Appendix H) do not need to provide analyses to demonstrate that these screening levels are ALARA. In addition, if residual radioactivity cannot be detected, it may be assumed that it has been reduced to levels that are ALARA. An ALARA analysis is also unnecessary in cases where soil removal is performed.[1] Therefore, the licensee may not need to conduct an explicit analysis to meet the ALARA requirement.

After a site has been decommissioned and the license terminated in accordance with the criteria in 10 CFR Part 20, "Standards for Protection against Radiation, Subpart E, "Radiological Criteria for License Termination," or after part of a facility or site has been released for unrestricted use in accordance with 10 CFR 50.83 and in accordance with the criteria in Subpart E, the Commission will require additional cleanup only if based on new information, it determines that the criteria of this subpart were not met and residual radioactivity remaining at the site could result in significant threat to public health and safety. Because ALARA represents an optimization technique to reduce doses below the dose criteria, it is not considered reasonable to reopen consideration of a previously released radiological area that meets the appropriate dose criterion.

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

- **The method is simple**. The method for most licensee applications should be simple, because the effort needed for very sophisticated models cannot generally be justified. In

[1] See preamble to the license termination rule found at 62 FR 39058; July 21, 1997.

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

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

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

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

Issues raised in 72 FR 46102 that pertained to guidance in the previous revision to this volume (Appendix N in NUREG-1757, Volume 2, Revision 1) are addressed in Revision 2 to this appendix. Issues included guidance on (i) discounting rates and (ii) monetary value of collective dose averted among others.

## N.2 ALARA as it Applies to NRC Decommissioning Regulations

ALARA, as defined in 10 CFR 20.1003, means:

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

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

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

This is a general form of ALARA that conceptually applies to all radiation protection dose limits under 10 CFR Part 20, as confirmed in *Shieldalloy Metallurgical Corp.* (Decommissioning of the Newfield, New Jersey Site), CLI-11-12, 74 NRC 460, 480 (Oct. 12, 2011), which states that "ALARA is a general requirement for all 'doses to members of the public' established in the 'Radiation Protection Programs' in 10 C.F.R. Part 20, including the license termination dose criteria.[2]" As such, all licensees should establish programs and controls with ALARA concepts in mind to reduce occupational doses and doses to members of the public to ALARA levels.

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

The licensee's implementation of the general ALARA principle from 10 CFR 20.1101(b) should be appropriate for the specific regulatory basis being utilized, which could include 10 CFR 20.1402 ("Radiological Criteria for Unrestricted Use"), 10 CFR 20.1403 ("Criteria for License Termination Under Restricted Conditions"), or 10 CFR 20.1404 (Alternate Criteria for License Termination) for unrestricted release, restricted release, or alternative release criteria, respectively. The following sections discuss general ALARA considerations for each of these regulatory bases.

### N.2.1 ALARA in Unrestricted Use

As described in the previous section, ALARA is a requirement of 10 CFR 20.1402. Thus, for most instances of unrestricted release, the ALARA requirement is to consider reasonably

[2] The applicability of 10 CFR 20.1101(b) to the 10 CFR Part 20, Subpart E, license termination dose criteria is discussed in Commission Order *Shieldalloy*, CLI-11-12, 74 NRC 460, (480-481). This order also provides additional details on the specific application of the ALARA regulation to the license termination rule.

achievable means of reducing dose below the 0.25 mSv (25 mrem) per year dose criterion of 10 CFR 20.1402. 10 CFR 20.1402 indicates that a site will be considered acceptable for unrestricted use if residual radioactivity meets the 0.25 mSv per year dose standard "and that the residual radioactivity has been reduced to levels that are as low as reasonably achievable (ALARA)". 10 CFR 20.1402 goes on to state that 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.

### N.2.2 ALARA in Restricted Release

Under 10 CFR 20.1403, a licensee must meet certain eligibility criteria before the NRC can terminate a license under restricted use conditions. Among other things, under 10 CFR 20.1403(a) a site will be considered acceptable for license termination under restricted conditions if the licensee can demonstrate that further reductions in residual radioactivity[3] necessary to comply with the provisions of 10 CFR 20.1402 would result in net public or environmental harm or were not being made because the residual levels associated with restricted conditions are ALARA.

This eligibility determination implements the NRC's preference for licensees to decommission to the unrestricted use criteria. The ALARA principle incorporated into § 20.1403(a) serves as a regulatory tool to limit the use of restricted release—effectively, to screen out sites that should be removing contamination to achieve unrestricted use. The NRC expects licensees to make every reasonable effort to achieve unrestricted release. Specifically, the requirement calls for a licensee seeking to use restricted release to analyze whether it would be cost-beneficial to remove enough radioactive contamination from the site so that doses to the public are no higher than 0.25 mSv/y (25 mrem/y) without reliance on restricted release controls. Such reduction of radioactivity is accomplished through removal of radioactive material or site decontamination. There are generally two alternative analyses that may be used to weigh the costs and benefits of removing radioactive contamination: (1) compare the potential benefits to the potential costs that are typically evaluated in an ALARA analysis, or (2) consider the net public and environmental harm as a cost and compare those costs against the health and environment-related benefits of removing radioactive contamination. Sections N.3–N.6 contain details on the implementation of these two analyses.

#### *N.2.2.1 Restricted Release—ALARA with Institutional Controls and without*

Once the eligibility for restricted release has been established under 10 CFR 20.1403(a), the general ALARA requirement still applies to the other criteria of 10 CFR 20.1403 for restricted release. Specifically, 10 CFR 20.1403(b) and 20.1403(e) enumerate dose limits for restricted release with institutional controls in place and without, respectively. Despite the availability of institutional controls and engineered barriers to reduce dose under restricted release, licensees must reduce residual radioactivity to levels that provide reasonable assurance that doses will not exceed a maximum value or "cap" specified in paragraph (e) if institutional controls are no longer in effect. As stated above, reduction of radioactivity is accomplished through removal of radioactive material or site decontamination.

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3 As noted in *Shieldalloy Metallurgical Corp.* (Decommissioning of the Newfield, New Jersey Site), CLI-13-6, 78 NRC 155, 168-170 (August 5, 2013), such reduction of radioactivity is only accomplished through removal of radioactive material, and the eligibility for restricted release must consider whether further reductions, without considering the impacts of institutional controls and engineered barriers, are cost beneficial.

Paragraph (b) requires provisions for legally enforceable institutional controls that provide reasonable assurance that the TEDE from residual radioactivity distinguishable from background to the average member of the critical group will not exceed 0.25 mSv (25 mrem) per year. Although not specifically included in paragraph (b), the general ALARA requirement applies and the dose from residual radioactivity must be as low as reasonably achievable even with institutional controls in place.

Paragraph (e) requires, in part, that

> Residual radioactivity at the site has been reduced so that if the institutional controls were no longer in effect, there is reasonable assurance that the TEDE from residual radioactivity distinguishable from background to the average member of the critical group is as low as reasonably achievable and would not exceed either—
>
> (1) 1.0 mSv (100 mrem) per year; or
>
> (2) 500 mrem (5 mSv) per year provided the licensee—
>
> (i) Demonstrates that further reductions in residual radioactivity necessary to comply with the 1.0 mSv/y (100 mrem/y) value of paragraph (e)(1) of this section are not technically achievable, would be prohibitively expensive, or would result in net public or environmental harm.

As the underlined portions in the preceding text indicate, paragraph (e) includes an explicit reference to ALARA and a requirement for an additional analysis similar to one for ALARA in 10 CFR 20.1403(a). First, the licensee must provide reasonable assurance that through reduction of residual radioactivity that the TEDE if institutional controls fail is ALARA. Second, to use the larger 5 mSv (500 mrem) per year cap, licensees must make an additional demonstration using criteria similar to those for 20.1403(a). For subparagraph (e)(2)(i), however, the requirement is not simply to reduce to as low as reasonably achievable, but to show that meeting the dose limit in paragraph (e)(1) is not "technically achievable, would be prohibitively expensive, or would result in net public or environmental harm." Further, 10 CFR 20.1403(e)(2) is a regulatory tool to limit the use of the 500 mrem (5 mSv) per year cap, similar to the way that the ALARA requirement in 10 CFR 20.1403(a) limits the use of restricted release.

### N.2.3 ALARA in Alternate Release Criteria

Under 10 CFR 20.1404, the Commission may terminate a license using alternate criteria greater than the dose criterion of 10 CFR 20.1402, 20.1403(b), and 20.1403(d)(1)(i)(A), if the licensee meets certain conditions. That is, the NRC might authorize release of a site with a TEDE to the average member of the critical group from residual radioactivity distinguishable from background in excess of 25 mrem (0.25 mSv) per year under the conditions specified in 10 CFR 20.1404. The general ALARA requirement of 10 CFR 20.1101(b) continues to apply even if an alternate dose criterion is authorized under 10 CFR 20.1404. Additionally, the criteria for approval of alternate release criteria includes specific ALARA requirements: the licensee must employ, to the extent practical, restrictions on site use according to the provisions of § 20.1403 in minimizing exposures at the site; and must reduce doses to ALARA levels, taking into consideration any detriments such as traffic accidents expected to potentially result from decontamination and waste disposal.

Based on these criteria, the NRC expects the licensee to consider requesting alternate release criteria only after first considering unrestricted and restricted release, and after implementing the provisions of 10 CFR 20.1403 on restrictions of site use to the extent practical to minimize dose.

### N.2.4 ALARA Measures and Radon

One area where the implementation of ALARA principles notably differs between unrestricted, restricted, and restricted release/alternate release criteria is that of radon mitigation:

> Because of [the] variations and the limitation[s] of measurement techniques, the Commission believes that it is not practical for licensees to distinguish between radon from licensed activities at a dose comparable to a 0.25 mSv/y (25 mrem/y) dose criterion and radon which occurs naturally. Therefore, in implementing the final rule, licensees will not be expected to demonstrate that radon from licensed activities is indistinguishable from background on a site-specific basis. Instead this may be considered to have been demonstrated on a generic basis when radium, the principal precursor to radon, meets the requirements for unrestricted release, without including doses from the radon pathway. (62 FR 39083)

Therefore, for unrestricted release, the NRC does not require licensees to include doses from the radon pathway as long as residual radium contamination is within the unrestricted use criteria. However, for restricted release:

> …it may not be reasonable to achieve levels of residual concentrations of radon precursors within the limit for unrestricted use…[F]or cases such as these, restricting site use by use of institutional controls could be considered by a licensee as a means to limit the doses from precursors by limiting access to the site. Under the restricted use provisions of the rule, these doses are required to be further reduced based on ALARA principles. (62 FR 39083)

For these reasons, licensees may need to consider institutional controls to reduce doses from radon, and its precursors, to comply with the ALARA principles of 10 CFR 20.1403. ALARA considerations should address the practicality of radon mitigation techniques in structures as part of institutional controls. ALARA considerations for compliance with the general ALARA principle as applied to 10 CFR 20.1403(e) for controls not in place do not have to address the radon pathway.

## N.3 Evaluation of Cost-Benefit ALARA Analyses

As discussed in Section N.1, compliance with the license termination criteria must include a demonstration that the dose to the average member of the critical group and the amount of residual radioactivity remaining on the site at license termination is ALARA. A simplified method for demonstrating compliance with the ALARA requirement is described below. Licensees may use more complex or site-specific analyses if more appropriate for their specific situation. In general, more complex analyses should follow the general concepts presented here. Evaluation of more complex analyses should be handled on a case-by-case basis and early involvement of the appropriate regulatory agencies and members of the public is suggested.

Sometimes it is very difficult or impossible to place a monetary value on an impact. A best effort should be made to assign a monetary value to the impact, because there may be no other way to compare benefits to costs. However, there may be situations for which a credible monetary

value cannot be developed. In these situations, a qualitative treatment may be the most appropriate. Qualitative analyses should be evaluated on their merits on a case-by-case basis.

### N.3.1 Simplified Method for Cost-Benefit ALARA Analyses

The simplified method presented here is to estimate when a proposed remediation action is ALARA using generalized estimates for the remedial action. Evaluating whether a proposed action is ALARA involves considering potential further, alternative remedial actions that would result in further reduction in the dose to the average member of the critical group or further reduction in the amount of residual radioactivity that would remain on site at license termination, relative to the proposed action. The evaluation then involves comparing the proposed action and the potential alternative remedial actions in terms of desired beneficial effects (benefits) and undesirable effects (costs). If the proposed action provides the greatest benefits: cost ratio when compared to a further alternative action, then that is demonstration that the proposed action is ALARA in comparison to the alternative action and the alternative action is unnecessary. Conversely, if the alternative action provides the greatest benefits: cost ratio, it would be ALARA to select the alternative option.

The various ALARA evaluations for license termination may all use this same simplified method, but some of the evaluations should differ in the nature of the baseline and alternative approaches being compared. The NRC staff should consider, in particular, whether the approaches being compared appropriately address the reduction of dose to the average member of the critical group or reduction of residual radioactivity remaining on site at license termination. For requests to use alternative release criteria under 10 CFR 20.1404, the NRC staff should evaluate the ALARA demonstrations on a case-by-case basis.

To compare the benefits and costs of a remediation action, it is necessary to use a comparable unit of measure. The unit of measure used here is the dollar; if possible, all benefits and costs are given a monetary value. The licensee can calculate benefits and costs as described in Sections N.2 through N.6.

The licensee should apply the method during remediation planning, before the start of remediation but after completing some or all of the characterization work. The licensee should only use the method to determine whether and where it should take particular remediation actions to meet the ALARA requirement.

If the licensee has already decided to perform a remediation action, there is no need to analyze whether the action was necessary to meet the ALARA requirement. The analysis described in this section is needed only to justify *not* taking a further remediation action. For example, if a licensee plans to wash room surfaces (either to meet the dose limit or as a good practice), there is no need to analyze whether the remediation action of washing is necessary to meet the ALARA requirement.

Table N.1 gives an example of various benefits and costs. Other than Collective Dose Averted, the additional benefits listed are generally only important in comparisons between alternatives that address whether the licensee can pursue restricted release. The value of any benefit or cost can be negative in some cases.

**Table N.1 Possible Benefits and Costs Related to Decommissioning**

| Possible Benefits | Possible Costs |
|---|---|
| Collective Dose Averted | Remediation Costs (including waste disposal costs) |
| Regulatory Costs Avoided | Additional Occupational/Public Dose |
| Changes in Land Values | Occupational Non-radiological Risks |
| Esthetics | Transportation Direct Costs and Implied Risks |
| Reduction in Public Opposition | Environmental Impacts |
| | Loss of Economic Use of Site/Facility |

## N.3.2 Calculation of Benefits

### *N.3.2.1 Collective Dose Averted*

In the simplest form of the analysis, the only benefit estimated from a reduction in the level of residual radioactivity is the monetary value of the collective averted dose to future occupants of the site. The licensee should base the collective averted dose on the same exposure scenario(s) used for the compliance calculations. Section N.2.7 discusses additional considerations related to groundwater residual radioactivity.

To calculate the benefit from collective averted dose, $B_{AD}$, determine the present worth of the future collective averted dose and multiply it by a factor to convert the dose to monetary value, as shown in Equation N-1 below.

NUREG/BR-0058, "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission," Revision 4, issued September 2004 describes present-worth calculations using both 3-percent and 7-percent real discount rates. Special considerations arise when comparing benefits and costs across generations, as is often the case with decommissioning. Thus, ethical and technical arguments can also support the use of lower discount rates. Therefore, if licensees anticipate important intergenerational consequences, such as for cases with radionuclides with half-lives of decades or longer, licensees should consider supplementing the analysis with an explicit discussion of the intergenerational concerns, such as how future generations will be affected by the regulatory decisions. Supplemental information could include a presentation of the values and impacts at the time they are incurred, with no present worth conversion. Also, licensees should consider a sensitivity analysis using a lower, but positive, discount rate. As a general principle, licensees should perform a sensitivity or uncertainty analysis, or both, whenever the values of key attributes can range widely. NUREG/BR-0058 contains additional discussion on the consideration of intergenerational consequences and sensitivity or uncertainty analyses.

$$B_{AD} = V_{AD} \times PW(AD_{Collective}) \quad \text{(N-1)}$$

where $B_{AD}$ = benefit from an averted dose for a remediation action, in current U.S. dollars

$V_{AD}$ = value of averted dose, which is a conversion factor for the monetary value of radiation dose (dollars ($) per person-rem, see NUREG/BR-0058). This conversion factor is applied to averted

dose (as in this case) and to dose incurred (as in the case of dose to workers or the public discussed in Section N.2.3).

$PW(AD_{collective})$ = present worth of a future collective averted dose in person-rem

The value of averted dose is periodically reviewed based on changes to the underlying assumptions. It is important to verify the current value from NRC regulatory guidance when performing a collective dose averted analysis. In addition to NUREG/BR–0058, licensees should consult NUREG-1530, "Reassessment of NRC's Dollar per Person-Rem Conversion Factor Policy".

The present worth of the future collective averted dose can be estimated from Equation N–2, for relatively simple situations:

$$PW(AD_{Collective}) = P_D \times A \times 0.025 \times F \times \frac{Conc}{DCGL_W} \times \frac{1 - e^{-(r+\lambda)N}}{r+\lambda} \quad \text{(N-2)}$$

where $P_D$ = population density for the critical group scenario in people/m$^2$;
$A$ = area being evaluated in square meters (m$^2$);
0.025 = annual dose to an average member of the critical group from residual radioactivity at the derived concentration guideline level ($DCGL_W$) concentration in rem/y;
$F$ = effectiveness, or fraction of the residual radioactivity removed by the remediation action;
$Conc$ = average concentration of residual radioactivity in the area being evaluated in units of activity per unit area for buildings or activity per unit volume for soil;
$DCGL_W$= derived concentration guideline equivalent to the average concentration of residual radioactivity that would give a dose of 0.25 mSv/y (25 mrem/y) to the average member of the critical group, in the same units as *"Conc"*;
$r$ = monetary discount rate in units per year;
$\lambda$ = radiological decay constant for the radionuclide in units per year; and
$N$ = number of years over which the collective dose will be calculated.

NOTE: When the discount rate, $r$, is zero and the radiological decay rate, $\lambda$, is very small so that $r + \lambda \rightarrow 0$, this equation must be adjusted (see Section N.6).

The present worth of the benefit calculated by Equation N–2, above, assumes that the peak dose occurs in the first year. This is almost always true for the building occupancy scenario but not always true for the residential scenario, where the peak dose can occur in later years. In that case, Equation N–2 would overestimate the benefit. The licensee may perform a more exact calculation that avoids this overestimation of the benefit of remediation by calculating the dose during each year of the evaluation period and then calculating the present worth of each year's dose. Section N.6 contains a detailed derivation of Equation N-2 and some of the other equations.

The $DCGL_W$ (derived concentration guideline level, average concentrations over a wide area) is based on compliance with the 0.25 mSv/y (25 mrem/y) dose limit. Base the population density, $P_D$, on the dose scenario used to demonstrate compliance with the dose limit. Thus, for buildings, the licensee should estimate $P_D$ for the building occupancy scenario. For soil, it

should base the $P_D$ on the residential scenario. The factor at the far right of the equation, which includes the exponential terms, accounts for both the present worth of the monetary value and radiological decay.

If more than one radionuclide is present, the total benefit from a collective averted dose, $B_{AD}$, is the sum of the collective averted dose for each radionuclide. When multiple radionuclides have a fixed concentration (i.e., secular equilibrium), residual radioactivity below the dose criteria is normally demonstrated by measuring one radionuclide and comparing its concentration to a $DCGL_W$ that has been calculated to account for the dose from the other radionuclides. In this case, the licensee may use the adjusted $DCGL_W$ with the concentration of the radionuclide being measured. The other case is where the ratio of the radionuclide concentrations is not fixed and varies from location to location within a survey unit; this benefit is the sum of the collective averted dose from each.

*N.3.2.2 Regulatory Costs Avoided*

This benefit usually occurs in ALARA analyses of restricted release versus unrestricted release decommissioning goals. By releasing the site with no restrictions, the licensee may avoid the various costs associated with restricted release. These costs can include

- additional licensing fees for safety reviews and for developing an Environmental Impact Statement,
- financial assurance for necessary control and maintenance of a site (10 CFR 20.1403(c)),
- costs (including NRC-related) associated with public meetings or the community review committee (10 CFR 20.1403(d)(2)), and
- future liability.

When evaluating the eligibility of a licensee's proposal for restricted release according to 10 CFR 20.1403(a), the NRC staff recommends that the regulatory costs avoided be included in the benefits of the unrestricted release decommissioning alternative, rather than included as costs related to the restricted release (see Section N.2.2).

*N.3.2.3 Changes in Land Values*

The licensee should account for any expected change in the value of the site or facility or surrounding land caused by the different decommissioning options. This may be difficult to quantify.

*N.3.2.4 Esthetics/Reduction in Public Opposition*

These can be very difficult to quantify. The licensee may wish to evaluate the effect of the available decommissioning options with respect to the overall esthetics (including the decommissioning activities themselves) of the site and surrounding area. Another factor the licensee may wish to consider is the potential reduction in opposition, if there is any, to the decommissioning activities or goal it is attempting to propose.

### N.3.3 Calculation of Costs

The licensee should evaluate the costs of the selected alternative remediation actions being evaluated. When performing a fairly simple evaluation, the costs generally include the monetary costs of

- the remediation action being evaluated,
- transportation and disposal of the waste generated by the action,
- workplace accidents that occur because of the remediation action,
- traffic fatalities resulting from transporting the waste generated by the action,
- doses received by workers performing the remediation action, and
- doses to the public from excavation, transport, and disposal of the waste. The licensee may also include other costs that are appropriate for the specific case.

The total cost, $Cost_T$, which is balanced against the benefits, has several components.

$$Cost_T = Cost_R + Cost_{WD} + Cost_{ACC} + Cost_{TF} + Cost_{WDose} + Cost_{PDose} + Cost_{other} \quad \text{(N-3)}$$

where $Cost_R$ = monetary cost of the remediation action (may include "mobilization" costs);
$Cost_{WD}$ = monetary cost for transport and disposal of the waste generated by the action;
$Cost_{ACC}$ = monetary cost of worker accidents during the remediation action;
$Cost_{TF}$ = monetary cost of traffic fatalities during transporting of the waste;
$Cost_{WDose}$ = monetary cost of dose received by workers performing the remediation action and transporting waste to the disposal facility;
$Cost_{PDose}$ = monetary cost of the dose to the public from excavation, transport, and disposal of the waste; and
$Cost_{other}$ = other costs as appropriate for the particular situation.

The licensee does not necessarily have to calculate all the costs described below. For example, if one or two of the costs can be shown to be in excess of the benefit and if none of the other costs are negative, the remediation action will be unnecessary. However, some of these costs may, in fact, be negative (i.e., the alternative may cost less than the preferred option) in some comparisons between alternative decommissioning options, and thus, the licensee may need to evaluate all costs.

#### *N.3.3.1 Remedial Action Costs*

Calculation of the incremental remedial action costs includes the standard manpower and mechanical costs. The licensee can account for any additional licensing fees from the NRC (e.g., if the option to meet the ALARA goal requires another year of remediation). Lower concentrations may change sampling or survey requirements. The remedial action can consider increased survey costs but note that this is the incremental cost of surveying below the

dose limit. Survey costs related to evaluating compliance at the dose limit are not part of the ALARA analysis.

#### *N.3.3.2 Transport and Disposal of the Waste*

The cost of waste transport and disposal, $Cost_{WD}$, may be evaluated according to Equation N-4.

$$Cost_{WD} = V_A \times Cost_V \quad \text{(N-4)}$$

where: $V_A$ = volume of waste produced, remediated in units of $m^3$; and
$Cost_V$ = cost of waste disposal per unit volume, including transportation cost, in units of $/$m^3$.

#### *N.3.3.3 Non-radiological Risks*

The cost of non-radiological workplace accidents, $Cost_{ACC}$, may be evaluated using Equation N-5.

$$Cost_{ACC} = V_{SL} \times F_W \times T_A \quad \text{(N-5)}$$

where: $V_{SL}$ = monetary value of a statistical life (or fatality) (see NUREG-1530, "Reassessment of NRC's Dollar per Person-Rem Conversion Factor Policy")—this value is subject to periodic revision, so it is important to verify the current value from NUREG-1530 when performing an analysis on non-radiological workplace accidents;
$F_W$ = workplace fatality rate in fatalities/hour worked; and
$T_A$ = worker time required for remediation in units of worker-hours.

#### *N.3.3.4 Transportation Risks*

Calculate the cost of traffic fatalities incurred during the transportation of waste, $Cost_{TF}$, as in Equation N-6.

$$Cost_{TF} = V_{SL} \times \left(\frac{V_A}{V_{SHIP}}\right) \times F_T \times D_T \quad \text{(N-6)}$$

where: $V_{SL}$ = monetary value of a statistical life (or fatality) (see NUREG-1530, "Reassessment of NRC's Dollar per Person-Rem Conversion Factor Policy")—this value is subject to periodic revision, so it is important to verify the current value from NUREG-1530 when performing an analysis on non-radiological workplace accidents;
$V_A$ = volume of waste produced in units of $m^3$;
$F_T$ = fatality rate per truck-kilometer traveled in units of fatalities/truck-km;
$D_T$ = distance traveled in km; and
$V_{SHIP}$ = volume of a truck shipment in $m^3$.

The actual parameters should depend on the site's planned method of waste transport. Some facilities may consider a mix of trucking and rail transport to take the waste to the disposal site.

In these cases, the cost would be equivalent to the total fatalities likely from the rail transport and the limited trucking, not just the trucking alone.

#### *N.3.3.5 Worker Dose Estimates*

Calculate the cost of the remediation worker dose, $Cost_{WDose}$, as shown in Equation N-7.

$$Cost_{WDose} = V_{AD} \times D_R \times T \quad \text{(N-7)}$$

where:
- $V_{AD}$ = value of incurred dose, which is a conversion factor for the monetary value of radiation dose (dollars ($) per person-rem, see NUREG/BR-0058);
- $D_R$ = TEDE rate to remediation workers in units of rems/hr; and
- $T$ = time worked (site labor) to remediate the area in units of person-hour.

The cost of worker dose usually should not be discounted, because the dose is all incurred close to the time of license termination.

#### *N.3.3.6 Loss of Economic Use of Property*

A cost in the "other" category could include the fair market rental value or economic use for the site during the time the additional remediation work is being performed. These costs are usually associated with locations such as laboratories, hospital rooms, and industrial sites. This cost may be added to the costs in Equation N-3.

#### *N.3.3.7 Environmental Impacts*

Another cost that could fall into the other category would be a remediation action that could damage an ecologically valuable area or cause some other adverse environmental impact. Include these impacts as costs of the remediation action.

*N.3.3.8 Default Parameters*

**Table N.2 Acceptable Parameter Values for Use in ALARA Analyses** *(Shows Acceptable Values for some of the Parameters used in Performing These Calculations)*

| Parameter | Value | Reference and Comments |
|---|---|---|
| Workplace accident fatality rate, $F_W$ | $4.2 \times 10^{-8}$/hr | NUREG-1496, "Final Generic Environmental Impact Statement in Support of Rulemaking on Radiological Criteria for License Termination of NRC-Licensed Nuclear Facilities," and NUREG-1496, Volume 2, Appendix B, Table A.1, July 1997 |
| Transportation fatal accident rate, $F_T$ | Trucks: $3.8 \times 10^{-8}$/km | NUREG-1496, Volume 2, Appendix B, Table A.1 |
| Value of averted dose, $V_{AD}$, and value of statistical life, $V_{SL}$ | values are updated periodically | NUREG/BR-0058, NUREG-1530. It is important to verify the current values (see Section N.2.2) |
| Monetary discount rate, *r* | 0.03/y and 0.07/y discount rates, with special considerations for intergenerational consequences (as discussed in Section N.2.2) | NUREG/BR-0058 |
| Number of years of exposure, *N* | Buildings: 70 years<br>Soil: 1,000 years | NUREG-1496, Volume 2, Appendix B, Table A.1 |
| Population density, $P_D$ | Building: 0.09 person/$m^2$<br>Land: 0.0004 person/$m^2$ | NUREG-1496, Volume 2, Appendix B, Table A.1 |
| Excavation, monitoring, packaging, and handling soil | Soil: 1.62 person-hours/$m^3$ of soil | NUREG-1496, Volume 2, Appendix B, Table A.1 |
| Waste shipment volume, $V_{SHIP}$ | Truck: 13.6 $m^3$/shipment | NUREG-1496, Volume 2, Appendix B, Table A.1 |

## N.3.4 Residual Radioactivity Levels that are ALARA for Unrestricted Use

The residual radioactivity level that is ALARA for license termination for unrestricted use is the concentration, $Conc_{ALARA}$ at which the benefit from removal equals the cost of removal. If the total cost, $Cost_T$, is set equal to the present worth of the collective dose averted in Equation N-2, the ratio of the concentration, $Conc_{ALARA}$, to the $DCGL_W$ can be determined (derivation shown in Section N.7).

$$\frac{Conc_{ALARA}}{DCGL_W} = \frac{Cost_T}{V_{AD} \times P_D \times 0.025 \times F \times A} \times \frac{r + \lambda}{1 - e^{-(r+\lambda)N}} \quad \text{(N-8)}$$

All the terms in Equation N-8 are as defined previously.

Since $P_D$, $N$, and $r$ are constants that have generic values for all locations on the site, the licensee only needs to determine the total cost, $Cost_T$, and the effectiveness, $F$, for a specific remediation action. If the concentration at a location exceeds $Conc_{ALARA}$, it may be cost effective to remediate the location by a method where the total cost is $Cost_T$. Note that the concentration, $Conc_{ALARA}$, that is ALARA can be higher or lower (more or less stringent) than the $DCGL_W$, although licensees should meet the $DCGL_W$.

For cases of license termination for restricted use, the NRC staff and licensees should modify the equations derived in Section N.6 to include the additional benefits that are only applicable to comparisons of unrestricted and restricted use (see Section N.1).

### N.3.5 Examples of Calculations

*Example 1: Washing Building Surfaces*

This example considers a building with cesium-137 residual radioactivity (λ = 0.023/y). The remediation action to be considered is washing a floor of 100 $m^2$ area. The licensee estimates that this may cost $400 and may remove 20 percent ($F$ = 0.2) of the residual radioactivity. The most recent value of $V_{AD}$, at the time of this revision, is $5100 per person-rem. For buildings, generic parameters are: $P_D$ = 0.09 person/$m^2$, $r$ = 0.07/y, and $N$ = 70 years. Using these values in Equation N-8:

$$\frac{Conc_{ALARA}}{DCGL_W} = \frac{\$400}{\$5100 \times 0.2 \times 0.025 \times 0.09 \times 100} \times \frac{0.07 + 0.023}{1 - e^{-(0.07 + 0.023)70}} \quad \text{(N-9)}$$

$$\frac{Conc_{ALARA}}{DCGL_W} = 0.16 \quad \text{(N-10)}$$

To meet the ALARA requirement, the floor should be washed if the average concentration exceeds about 16 percent of the $DCGL_W$. This is more stringent than the dose limit. This calculation shows that washing building surfaces is often necessary to meet the ALARA requirement. If the surfaces are planned to be washed, there is no need for the licensee to perform the ALARA evaluation or to submit the evaluation to the NRC. If it is decided not to wash the building surfaces, the licensee could submit this evaluation and demonstrate in the FSS that all surfaces have a concentration below 16 percent of the $DCGL_W$.

*Example 2: Scabbling Concrete in a Building*

This example is the same as above, except that it evaluates the use of a scabbling tool that removes the top one-eighth of an inch of concrete. The licensee estimates that the total cost of the scabbling may be $5,000 for the 100 $m^2$ floor and that it may remove all the residual radioactivity so that $F$ = 1. Using these values in Equation N-8 gives:

$$\frac{Conc_{ALARA}}{DCGL_W} = \frac{\$5000}{\$5100 \times 1 \times 0.025 \times 0.09 \times 100} \times \frac{0.07 + 0.023}{1 - e^{-(0.07 + 0.023)70}} \quad \text{(N-11)}$$

$$\frac{Conc_{ALARA}}{DCGL_w} = 0.41 \quad \text{(N-12)}$$

The licensee could decide to scabble depending on the concentrations present. In lieu of scabbling, the licensee could provide this analysis and demonstrate that the floor concentration is less than 0.41 $DCGL_W$.

*Example 3: Removing Surface Soil*

In this example, soil with an area of 1,000 $m^2$ is found to contain cobalt-60 (λ = 0.1315/y) residual radioactivity to a depth of 15 centimeters (cm) (6 inches (in.)). The licensee estimates that the cost of removing the soil ($F$ = 1) may be $100,000. For soil, the generic parameters are $P_D$ = 0.0004 person/$m^2$, $r$ = 0.03/y, and $N$ = 1000 y. The most recent value of $V_{AD}$, at the time of this review, is $5100 per person-rem. Using these values in Equation N-8 gives:

$$\frac{Conc_{ALARA}}{DCGL_w} = \frac{\$100{,}000}{\$5100 \times 1 \times 0.025 \times 0.0004 \times 1000} \times \frac{0.03 + 0.1315}{1 - e^{-(0.03 + 0.1315)1000}} \quad \text{(N-13)}$$

$$\frac{Conc_{ALARA}}{DCGL_w} = 317 \quad \text{(N-14)}$$

Thus, meeting the dose limit would be limiting by a considerable margin. Based on these results, it would rarely be necessary to ship soil to a waste disposal facility to meet the ALARA requirement. The licensee could use this evaluation to justify not removing soil.

The advantage of the approach shown in these examples is that it allows the licensee to estimate a concentration at which a remediation action may be cost effective before starting remediation and before planning the FSS. Thus, it is a useful planning tool that lets the licensee determine which remediation actions may be needed to meet the ALARA requirement.

### N.3.6 When Mathematical Analyses are not Necessary

In certain circumstances, the results of an ALARA analysis are known on a generic basis and an analysis is not necessary. For residual radioactivity in soil at sites that may have unrestricted release, generic analyses (see NUREG-1496 and the examples in Section N.2.5) show that shipping soil to a licensed low-level waste disposal facility is unlikely to be cost effective for unrestricted release, largely because of the high costs of waste disposal. An ALARA analysis is not needed for soil removal to meet unrestricted release at or below a dose criterion of 0.25 mSv (25 mrem) per year.[4] Therefore, the licensee generally does not have to evaluate shipping soil to a low-level waste disposal facility to achieve exposure levels at or below the criterion for unrestricted release. However, this conclusion may not hold for waste disposal at other than licensed low-level waste disposal facilities.

In light of the conservatism in the building surface and surface soil default screening levels developed by the NRC staff (see Appendix H of this volume), the reviewer presumes, absent information to the contrary, that licensees who remediate building surfaces or surface soil to the NRC default screening levels do not need to provide analyses to demonstrate that these screening levels are ALARA.

[4] See preamble to the license termination rule found at 62 FR 39058; July 21, 1997.

In addition, the NRC staff concludes that licensees would not be required to perform a quantitative cost-benefit ALARA analysis for cases where no residual radioactivity distinguishable from background remains or will remain on the site at termination.

Removal of loose residual radioactivity from building surfaces is almost always cost effective, except when very small quantities of radioactivity are involved. Therefore, the NRC staff concludes that loose residual radioactivity normally should be removed from building surfaces and if it is removed, the cost-benefit analysis would not be needed.

### N.3.7 Additional Considerations for Residual Radioactivity in Groundwater

The method described above is adequate for most situations and has minimal cost for analyses. However, if the site has residual radioactivity from site operations in groundwater, the licensee should include other factors, as described below, if it intends to release the site.

If there is residual radioactivity from site operations in groundwater, it may be necessary to calculate the collective dose from consumption of the groundwater. Default or generic groundwater models typically assume that potable aquifers have small volumes and cannot supply large populations. When this is the case, dose calculations for the site critical group may adequately represent the collective dose from groundwater. However, when site-specific groundwater modeling is used, and the residual radioactivity is diluted in an aquifer of large volume and there is also an "existing population deriving its drinking water from a downstream supply using a downstream plume" (62 FR 39058; July 21, 1997), the collective dose for that population should be included in the ALARA calculation. The possibility of reducing the collective dose by remediation should be one of the items evaluated as one of the benefits, even if remediation would not affect the critical group's doses significantly. Another consideration for groundwater residual radioactivity would be the reduction of any potential costs incurred by other entities, such as a public water supply utility, to meet the requirements of the Safe Water Drinking Act, if the licensee's residual radioactivity levels would potentially lead to concentrations at the wellhead that would exceed the U.S. Environmental Protection Agency's Maximum Contaminant Levels.

## N.4 Determination of "Net Public or Environmental Harm"

Subpart E, 10 CFR 20.1403(a) and 10 CFR 20.1403(e)(2)(i) address circumstances in which a licensee may demonstrate that further remediation would cause net public or environmental harm. The calculation to demonstrate net public or environmental harm is a special case of the general ALARA calculation described above that compares the benefits in dose reduction to the cost of doses, injuries, and fatalities incurred or to the cost of environmental degradation. The calculation does not consider the monetary costs for performing further remediation, $Cost_R$, or the costs of waste disposal, $Cost_{WD}$. Thus, if the benefit from averted dose $B_{AD}$ is less than the sum of the costs of workplace accidents, $Cost_{ACC}$, the costs of transportation fatalities, $Cost_{TF}$, the costs of remediation worker dose, $Cost_{WDose}$, and the costs of public dose, $Cost_{PDose}$, or less than the costs of any environmental degradation, $Cost_{ED}$, then there is net public or environmental harm. Thus, there is net public or environmental harm if:

$$Net\ harm\ if: B_{AD} < Cost_{ACC} + Cost_{TF} + Cost_{WDose} + Cost_{PDose} \quad \text{(N-15)}$$

or

$$Net\ harm\ if: B_{AD} < Cost_{ED} \quad \text{(N-16)}$$

In some cases, it may be very difficult to assign a credible monetary value to environmental degradation. For example, environmental harm could be caused by an action such as remediation of a wetlands area. There may be no way to assign a monetary value to this action. In these cases, it is acceptable to use qualitative arguments, which should be evaluated on a case-by-case basis.

## N.5 Demonstration of "Not Technically Achievable"

Subpart E, 10 CFR 20.1403(e)(2)(i), addresses circumstances in which a licensee would be required to demonstrate that further reductions in residual radioactivity are not technically achievable. Remediation of residual radioactivity is almost always technically achievable, even if not economically feasible. This provision allows for special cases that may not be foreseeable; thus, the NRC has no specific guidance on this provision. Instead, the NRC staff will evaluate licensee submittals on a case-by-case basis.

## N.6 Demonstration of "Prohibitively Expensive"

Subpart E, 10 CFR 20.1403(e)(2), addresses circumstances in which a licensee would be required to demonstrate that further reductions in residual radioactivity would be prohibitively expensive. The licensee can demonstrate this by an analysis like the ALARA analysis described above but using an increased value of averted dose, $V_{AD}$. As discussed in Section N.2.2, NUREG-1530 and NUREG/BR-0058 should be consulted to determine the recommended value for averted dose. For a "prohibitively expense" assessment, this value should be multiplied times 10 prior to being used as $V_{AD}$ in the analysis. This increased value of averted dose reflects the statement in the final rule on radiological criteria for license termination that the NRC considers it appropriate that a remediation would be prohibitively expensive if the cost to avert dose were an order of magnitude more expensive than the cost recommended by the NRC for an ALARA analysis (see 62 FR 39058, p. 39071, July 21, 1997). However, the NRC also stated that "...a lower factor may be appropriate in specific situations when the licensee could become financially incapable of carrying out decommissioning safely." Thus, values lower than 10 times the value recommended in NUREG-1530 and NUREG/BR-0058 for $V_{AD}$ may be used when remediation actions could otherwise cause the licensee to become financially incapable of safely carrying out decommissioning.

## N.7 Derivation of Main Equations To Calculate ALARA Concentrations for Unrestricted Use

The following derivation applies to an ALARA evaluation for license termination for unrestricted use. Additional benefits would apply to restricted use cases (see Sections N.2.1 and N.2.2), and these equations would be modified.

The ALARA analysis compares the monetary value of the desirable effects (benefits) of a remediation action (e.g., the monetary benefit of collective averted dose) with the monetary value of the undesirable effects (e.g., the costs of waste disposal). If the benefits of a remediation action would exceed the costs, the licensee should take remediation action to meet the ALARA requirement.

The primary benefit from a remediation action is the collective dose averted in the future (i.e., the sum over time of the annual doses received by the exposed population). Assume:

If benefits > costs, the remediation action should be taken (N-17)

(1) A site proposes remediation and termination for unrestricted use (thus, the benefit to be considered is dose averted only).

(2) The site has an area with residual radioactivity at concentration, *Conc*.

(3) The concentration equivalent to 0.25 mSv/y (25 mrem/y) ($DCGL_W$) for the site has been determined (for soil or for building surfaces, as appropriate).

(4) The residual radioactivity at a site has been adequately characterized so that the effectiveness of a remediation action can be estimated in terms of the fraction *F* of the residual radioactivity that the action may remove.

(5) The peak dose rate occurs at time 0 and decreases thereafter by radiological decay.

The derived concentration guideline ($DCGL_W$) is the concentration of residual radioactivity that would result in a TED E to an average member of the critical group of 0.25 mSv/y (25 mrem/y). Therefore, the annual dose *D* to the average member of the critical group from residual radioactivity at concentration *Conc* is:

$$D = 0.025 \left(\frac{rem}{yr}\right) \times \frac{Conc}{DCGL_W} \quad \text{(N-18)}$$

If a remediation action would remove a fraction, *F*, of the residual radioactivity present, the annual averted dose to the individual, $AD_{individual}$, is

$$AD_{individual} \left(\frac{rem}{yr-person}\right) = F \times 0.025 \left(\frac{rem}{yr}\right) \times \frac{Conc}{DCGL_w} \quad \text{(N-19)}$$

The annual collective averted dose, $AD_{collective}$, can be calculated by multiplying the individual averted dose, $AD_{individual}$, by the number of people expected to occupy the area, *A*, containing the residual radioactivity. The number of people in the area containing the residual radioactivity is the area, *A*, times the population density, $P_D$, for the site.

Thus:

$$AD_{collective} = F \times 0.025 \left(\frac{rem}{yr}\right) \times \frac{Conc}{DCGL_w} \times A \times P_D \quad \text{(N-20)}$$

The annual monetary benefit rate at time 0, $B_0$, from the averted collective dose in dollars per year can be calculated by multiplying the annual collective averted dose, $AD_{collective}$, by the value of averted dose, $V_{AD}$:

$$B_0 = V_{AD} \times F \times 0.025 \left(\frac{rem}{yr}\right) \times \frac{Conc}{DCGL_w} \times A \times P_D \quad \text{(N-21)}$$

The total monetary benefit of averted doses can be calculated by integrating the annual benefit over the exposure time in years, considering both the present worth of future benefits and radiological decay. It is OMB and NRC policy to use the present worth of both benefits and costs that occur in the future.

The equation for the present worth, $P_{WB}$, of a series of constant future annual benefits, $B$ (in dollars per year), for $N$ years at a monetary discount rate of $r$ (per year) using continuous compounding is:

$$PW_B = B \times \frac{e^{rN} - 1}{re^{rN}} \quad \text{(N-22)}$$

The continuous compounding form of the present worth equation is used, because it permits an easy formulation that includes radiological decay. If the annual benefit rate, $B$, is not constant but is decreasing from an original rate, $B_0$, because of radiological decay, the radiological decay rate acts like an additional discount rate that can be added to the monetary discount rate of decrease, so that the present worth of the annual benefits $P_{WB}$ becomes:

$$PW_B = B_O \times \frac{e^{(r+\lambda)N} - 1}{(r+\lambda)e^{(r+\lambda)N}} \quad \text{(N-23)}$$

Dividing the numerator and denominator of the right-hand term by $e^{(r+\lambda)N}$ yields:

$$PW_B = B_O \times \frac{1 - e^{-(r+\lambda)N}}{r+\lambda} \quad \text{(N-24)}$$

As $N \rightarrow \infty$, Equation N-24 has the limit:

$$PW_B = B_O \times \frac{1}{r+\lambda} \quad \text{(N-25)}$$

When the discount rate, $r$, is zero and the radiological decay rate is very small, so that $r + \lambda \rightarrow 0$, and Equation N-24 has the limit:

$$PW_B = B_O \times N \quad \text{(N-26)}$$

The total benefit from the collective averted dose, $B_{total}$, is the present worth of the annual benefits. $B_{total}$ can be calculated by combining Equations N-21 and N-24:

$$B_{total} = V_{AD} \times F \times 0.025 \times \frac{Conc}{DCGL_w} \times A \times P_D \times \frac{1 - e^{-(r+\lambda)N}}{r+\lambda} \quad \text{(N-27)}$$

Now consider the total cost of a remediation action, $Cost_T$. The costs included in $Cost_T$ are (1) the direct cost of the remediation action itself, $Cost_R$, (2) the cost of waste disposal including its shipping cost, $Cost_{WD}$, (3) the monetary costs of workplace accidents during the remediation, $Cost_{ACC}$, (4) the monetary costs of transportation accidents during the shipping of waste, $Cost_{TF}$, (5) the monetary value of the dose that remediation workers receive, $Cost_{WDose}$, (6) the monetary value of the dose to the public from the excavation, transport, and disposal of the waste, $Cost_{PDose}$, and (7) other costs, as appropriate, for the specific site, $Cost_{other}$. Thus,

$$Cost_T = Cost_R + Cost_{WD} + Cost_{ACC} + Cost_{TF} + Cost_{WDose} + Cost_{PDose} + Cost_{Other} \quad \text{(N-28)}$$

What is of interest in this derivation is the concentration, $Conc_{ALARA}$, at which the benefit, $B_{total}$, equals the total cost, $Cost_T$. Thus, in Equation N-27, $Cost_T$ can be substituted for $B_{total}$, and then Equation N-27 can be solved for the concentration, $Conc_{ALARA}$, relative to the $DCGL_W$, as in Equation N-29.

$$\frac{Conc_{ALARA}}{DCGL_W} = \frac{Cost_T}{V_{AD} \times F \times 0.025 \times P_D \times A} \times \frac{r + \lambda}{1 - e^{-(r+\lambda)N}} \quad \text{(N-29)}$$

The licensee can use Equation N-29 to determine the concentration in an area for which it should take a remediation action to meet the ALARA criterion. The equation appears complicated but can be solved in a few minutes with a hand-held calculator, and it only has to be done once for each type of remediation action at a site. $P_D$, $N$, and $r$ are constants. Section N.2.3 (Table N.2) gives generic values for $P_D$ and $N$, or they may be determined on a site-specific basis. The only site-specific information that the licensee needs is the total cost, $Cost_T$, and the effectiveness, $F$, for each remediation action being evaluated.

# APPENDIX O

# GUIDANCE FOR THE USE OF COMPOSITE SOIL SAMPLING FOR DEMONSTRATING COMPLIANCE WITH RADIOLOGICAL RELEASE CRITERIA

## O.1 Introduction

This appendix provides information on the potential need for composite sampling, as well as the technical basis and guidance on acceptable approaches that licensees could use to incorporate composite sampling strategies into their FSS.[1] In addition, this guidance includes appropriate uses of composite sampling for generating the data for other decommissioning site investigations and surveys including characterization surveys. This appendix is based on the report titled "Technical Bases and Guidance for the Use of Composite Soil Sampling for Demonstrating Compliance with Radiological Release Criteria," (Vitkus, 2012).

MARSSIM, Revision 1, provides guidance on the number of samples needed to have confidence in the survey results (constrain Type I and II error rates to acceptable levels), as well as considering the number of samples needed to ensure elevated areas between sample locations can be detected during scanning. For example, surface scanning specifications for Class 1 survey units include a required scan MDC ($MDC_{SCAN}$) that is a function of the sample spacing and the respective $DCGL_{EMC}$ that provides assurance that elevated areas of concern are identified through either sampling or scanning. If the required $MDC_{SCAN}$ cannot be met, the number of samples may need to be increased to reduce the area between samples to match the actual $MDC_{SCAN}$ (i.e., a greater number of samples and denser sample spacing would lead to a smaller area between samples; smaller areas generally have higher allowable $DCGL_{EMC}$s and therefore, a less restrictive $MDC_{SCAN}$). For land area survey units (soil), use of scanning to detect elevated areas relies upon the radionuclides of concern (ROCs) being gamma emitters and, for the case of non-gamma-emitting ROCs (i.e., hard-to-detect [HTD] ROCs), reliance upon a surrogate relationship. However, the coupling of sample spacing with a required $MDC_{SCAN}$ cannot specifically be followed when the scenario involves HTD radionuclides as the only ROCs or when a surrogate relationship cannot be established. In certain cases (e.g., only HTD radionuclides are present, HTD radionuclides with no surrogate relationship, average survey unit concentrations that are close to the $DCGL_W$ or sites with relatively high variability in residual radioactivity concentrations), the number of samples required may start to become cost prohibitive. In these cases, licensees may propose composite sampling to reduce the total number of samples that are analyzed and hence the analytical cost.

The FSS-related portions of this guidance follow MARSSIM processes for demonstrating compliance with radiological release criteria and the average allowable residual contamination levels. However, additional evaluations are necessary to guide the use of composite sampling within the MARSSIM framework for the specific case of ensuring that elevated areas of concern are addressed when HTD radionuclides are a primary consideration. Both Federal agency and academic resources are available for supplementing the MARSSIM radiological survey processes with composite sampling and meeting the challenges presented by HTD radionuclides. These resources, coupled with the experience from chemically contaminated sites, provide the bases for the general approaches included in this appendix, that are applicable to identifying HTD elevated areas.

[1] As of the publication date of this volume, limited guidance is available on use of composite sampling in radiological surveys. MARSSIM, Revision 1 (2000), recommends that, if an inordinate number of samples are required, then the DQOs should be revisited. Chapter 14 of NUREG 1505 briefly introduces the concept of composite sampling as a means to reduce the total number of samples requiring analysis (NRC, 1998).

## O.2 Advantages and Disadvantages of Composite Sampling

There are scenarios where it could be advantageous for licensees to apply a composite sampling approach for the MARSSIM-based FSS DQOs and the associated Data Life Cycle. These scenarios normally will involve a specific set of conditions where the approach could be beneficial. Generally, such conditions would involve situations where the analytical costs are high, required $MDC_{SCAN}$ in Class 1 survey units are difficult to achieve, and/or the presence of HTD radionuclides both increase analytical costs and reduce detection capability. Therefore, successful implementation of composite sampling requires a well-thought-out plan and is normally only beneficial when the conditions above exist. Table O.1 summarizes when composite sampling is advantageous. The advantages numbered 1, 2, and 4 would be applicable for an FSS, and all of the advantages listed are applicable to other survey types, such as site characterization. The disadvantages listed must also be considered and addressed in the planning and data life cycle. Table O.1 also summarizes the uses and considerations that the licensee should include in its survey plan to ensure that composite sampling would not interfere with the decision as to whether or not the radiological release criteria are satisfied.

## O.3 Composite Sample Plan Design

This section discusses the conditions under which a composite sampling approach is appropriate and would be considered advantageous. This section also discusses conditions that may affect decisions made during the DQA phase.

**Table O.1 Composite Sampling Overview**

| Advantages | Disadvantages |
|---|---|
| 1. Reduces analytical costs.<br>2. Provides a better estimate of mean concentration in the study area over an equivalent number of individual samples.<br>3. Identifies units that have the highest contaminant levels.<br>4. With an appropriately adjusted contaminant benchmark/investigation level, composite sampling can increase the ability to detect elevated areas by increasing the number of locations sampled. | 1. Information is lost on the individual sample increments that make up a composite. This loss of information is a concern when testing to determine if a ROC exceeds a threshold (e.g., a $DCGL_{EMC}$) over a specific area because of possible dilution to one or more increments with elevated activity concentrations by the other composite increments.<br>2. Cannot be used when action levels ($DCGL_W$s) are near analytical detection limits or the natural background concentration levels.<br>3. For nonhomogeneous contaminant distributions, temporal or spatial variability information is lost.<br>4. Cannot be used when integrity of individual sample values change as a result of compositing (e.g., loss of volatile contaminants, due to the physical compositing mechanism). |
| **Considerations for Applying Composite Sampling** | |
| 1. Useful when the size of the pattern or feature of interest, such as elevated areas, is smaller than the spacing between the statistically required random sampling locations.<br>2. User must account for potential introduction of large additional errors due to heterogeneous nature of the contaminant in the matrix, or the matrix itself.<br>3. Aliquots used to form the composite must be of equivalent weight/volume and the individual aliquots and the composite itself must be well homogenized.<br>4. Must account for the dilution factor when evaluating the result against a threshold, most commonly a $DCGL_{EMC}$. Necessitates a modified investigation level (MIL).<br>5. In most cases, the user must maintain the ability to retest individual samples (increments) making up the composite to retrieve potentially lost information.<br>6. Should not be used to establish surrogate ratios.<br>7. May not be advisable for Scenario B analyses. | |

## O.3.1 Uses, Considerations, and Limitations for Composite Sampling

### *O.3.1.1 Uses for Composite Sampling*

- Composite sampling may be used to estimate the mean concentration of a ROC for various decommissioning surveys including scoping, characterization, remedial action support, and final status surveys. Before selecting a composite sampling approach, an analyst should recognize that although an equivalent or better estimate of the mean may result from this approach, there is a loss of crucial information regarding the overall variability of the ROCs within the study area. For example,

  - 10 locations—numbered 1 through 10—are selected for random sampling from the study area.

  - The analytically determined concentrations in pCi/g at these 10 locations are as follows:

| Sample No. | Sample Value | Sample No. | Sample Value |
|---|---|---|---|
| 1 | 3.1 | 6 | 9.7 |
| 2 | 3.9 | 7 | 3.4 |
| 3 | 8.9 | 8 | 6.8 |
| 4 | 2.0 | 9 | 6.6 |
| 5 | 1.9 | 10 | 6.9 |

  - The calculated study area mean and standard deviation (sigma) would be calculated to be 5.3 ± 2.8.

  - Then, assume two composites are formed and analyzed: Composite 1 from the even numbered locations and Composite 2 from the odd numbered locations. Assuming equal aliquots, the composite concentrations would be 5.7 and 4.6, respectively. The study area mean and sigma would be calculated to be 5.2 ± 0.8.

  This example illustrates that the mean is accurately estimated with 2 composites; however, the actual study area variability is significantly underestimated. This factor will be an important consideration, should data collected during an earlier phase of the RSSI process be used for planning future investigations (e.g., the data are used to determine the number of samples required to ensure adequate power[2] for hypothesis tests).

- Compositing may be used to estimate the proportion of a population exhibiting a trait, such as the presence or absence of a specific ROC.

[2] Used in this context, power is the probability of rejecting the null hypothesis when it is false. For Scenario A, the power is the ability to find that a clean site is clean.

- Composite sampling may be used to classify survey areas as containing elevated areas or identify the spatial distributions of survey area ROC concentration levels, such as those parts of the site with the highest concentration levels.

- Composite sampling may be used when the contaminant $DCGL_W$, mean, variability, and/or decision error combinations result in a relative shift <1 or otherwise require an inordinate number of samples. The site may elect to use composite sampling to reduce the number of samples requiring analysis, yet still meet the sample number to adequately estimate the survey unit mean/median for the selected statistical test.

- Composite sampling may be used as a method to decrease sample spacing. This may occur when the ROC may be a low abundance or low-energy gamma emitter with a high $MDC_{SCAN}$ relative to the required $MDC_{SCAN}$ and hot spot or elevated area identification considerations then become the driver for sample spacing and the respective $DCGL_{EMC}$.

- Composite sampling may be used to reduce analytical costs associated with HTD radionuclides, which typically require more expensive wet chemistry.

- Composite sampling may be used during characterization or to provide additional FSS survey unit coverage for Class 2 and 3 areas to ensure proper classification of the unit. Because Class 2 and Class 3 survey units should not have residual contaminant concentrations in excess of the $DCGL_W$ when properly classified, under most FSS conditions, there is limited, if any, benefit to composite sampling in properly classified Class 2 or 3 FSS units. Use of composite sampling in these classifications would necessitate application of an MIL that is a fraction of the reclassification investigation level.

- Composite sampling may be used for HTD radionuclides for which an actual $MDC_{SCAN}$ cannot be established (e.g., pure beta or alpha emitter in soil) and there are no surrogate radionuclide relationships available. The composite sampling is used as a method to increase the probability of elevated area or hot spot detection and as a means to reduce analytical cost. However, this situation would require considerable evaluations performed on a case-by-case basis. As such, this guidance provides only a general scenario and the associated variables. If composite sampling is proposed to alleviate sampling requirements associated with HTD radionuclides, the licensee should contact the NRC early in the process to discuss the acceptability of the proposal.

- Composite sampling may be used when maintaining sample density is important, yet one of the user's objectives is to cover a larger area without increasing the analytical budget.

Each of the above applications will require more rigorous data assessments and may require retesting of the individual increments comprising a composite in certain cases. These case requirements are discussed individually in Section O.4.

*O.3.1.2 General Considerations*

The justification for incorporating composites into a sampling plan relies upon several factors. Factors that will assist in properly using composites are discussed below.

(1) Composite sample data may be applied without modification when information on individual samples is not important for the decisions that will be made with the data. When a threshold concentration or IL is of importance (e.g., the $DCGL_W$ for a Class 2 survey unit or a $DCGL_{EMC}$ for a Class 1 survey unit), then a MIL must be established. The MIL will be a fraction of the concentration threshold based on the number of increments (*k*) that comprise the composite. There are at least two options for setting the MIL limits. In most of the literature, the MIL is commonly defined as *IL/k*. Selection of the MIL value is critical. A too high MIL may result in missing discrete samples that exceed the IL (false negative). A too low MIL will conversely result in incorrectly investigating composite results that did not contain increments exceeding the IL (false positives). That is, a high false positive rate would be expected in cases where either *k* is too high for the site conditions, where there is not an expected substantial difference between the IL and the estimated site concentrations, or a combination of the two. The default MIL proposed under most conditions should be established:

$$MIL = \frac{IL}{k}$$

Revisions to the default MIL determination would require technical justification.[3] In any case, the site must provide both a lower and upper bound of the MIL for which definitive decisions may be made on a given result being definitely below or above a set action level.

(2) Composites are useful when analytical costs are high; otherwise, composite sampling is generally not likely to be cost effective. Furthermore, additional costs associated with forming composites and packaging and maintaining the increments is factored into the cost differential, before determining the value added of composite sampling.

(3) Composite sampling is properly used when it will not affect the analyte integrity; it should not be used, for example, for volatile analytes.

(4) The sample matrix must be amenable to homogenization, and each increment must be equally represented in the composite.

(5) Analytical detection limits and/or background interferences must be sufficiently low, relative to the proposed MIL, that the probability of misclassifying a composite sample result and obtaining a result less than the MIL is negligible. An example for illustration would be if the Sr-90 NRC screening level $DCGL_W$ of 1.1 pCi/g were the IL. A typical soil matrix analytical detection limit is 0.8 pCi/g. Composite sample results would be expected to have a high false negative rate and should not be used unless the objective of the study is to only identify those areas of a site exhibiting high concentrations, as Section O.4.5 discusses.

#### *O.3.1.3 Limitations*

Radiological survey plans that include composite sampling should be reviewed to ensure that the plan either addresses the following inherent limitations, or the limitation will not adversely affect the data decisions.

3 R. Correll (2001) described an approach that licensees may benefit from considering.

(1) Reduction in the information on variability. This limitation is potentially detrimental for situations such as calculating a sufficient number of samples that will satisfy statistical power requirements and DQOs for hypothesis testing, during survey planning and others.

(2) Due to loss of information on background variability, composite sampling may not be appropriate for Scenario B analyses. When attempting to show survey unit concentrations are indistinguishable from background (Scenario B analysis), background variability should be assessed and retrospective power should be determined to ensure a sufficiently low Type II (false negative) error rate (low error of retaining the null hypothesis that the site is clean when the site is dirty). This is made more difficult using a composite sampling strategy, which tends to underestimate variability.

(3) Potential loss of temporal or spatial information. In some scenarios, specific knowledge of the concentration in a unit area smaller than that represented by the composite is important in the decision process (e.g., for differentiating classifications of a site).

(4) Difficulty in homogenizing matrices. Such difficulty may be anticipated with clay soils and/or in cases where the contaminant is present as small particles that cannot be uniformly distributed.

(5) Lost information on maximum concentrations. This limitation is important for threshold investigations and may be counteracted with appropriate protocols for retesting the questionable composite sample's individual increments.

(6) Lost information on concentration correlations for two or more ROCs. As a result, composite sampling is not appropriate for determining surrogate relationships, such as estimating the surrogate correlations.

### O.3.2 Composite Sampling Plan and Data Quality Assessment Review Items

The following should be considered when reviewing composite sampling plans:

- The objective of the composite protocol, as well as how the resultant data will be assessed is described.

- Whether the composite sampling design is implemented following proper procedures/project specific instructions. American Standards and Testing Materials (ASTM D 6051-96, Reapproved 2006), "Standard Guide for Composite Sampling and Field Subsampling for Environmental Waste Management Activities," provides a potentially useful reference when developing procedures. Procedures should include the following:

    - Whether each composite increment is collected, and a representative aliquot from each increment is containerized for possible reanalysis.

    - Whether the remaining portions of each increment are homogenized and the composite sample containerized.

    - Whether each increment contributes an equivalent volume/weight to the composite.

- Whether the bases for the number of increments, *k*, per composite sample are provided. Several factors to be considered include, but are not limited to, the physical nature of the samples, the anticipated concentrations of the ROCs relative to the detection limits (to ensure that a single increment above the action level could be detected), and the capability to combine and homogenize the increments adequately. The larger *k* becomes, the more difficult it will be to adequately homogenize the increments and, therefore, an increase in the sampling error should be expected. To better control the sampling error, there should be a limit to the number of increments. The analytical detection limit (*d*) will also affect the value of *k* for a composite sample. This value is defined as:

$$k < IL/d \text{ (EPA 1995)}$$

  When determining the number of increments, *k*, that the ratio of the analytical cost to composite acquisition costs, and the estimated ratio between the "error variability" component of the sample collection and measurement processes and the "inherent concentration variability" component were considered. Guidance for optimal increment determination is provided in EPA's Quality System Document, QA/G-5S (EPA, 2002). Specific factors that should be considered include the following:

  - The analytical MDC relative to the $DCGL_W$ and/or MIL (i.e., the MDC should be lower than the MIL).

  - The ratio of the measurement error standard deviation (includes random error in sample collection and compositing and the measurement error) to the inherent variation standard deviation (variability in the concentration of the target population). The optimal value for *k* will decrease as this ratio approaches unity. With increasing values of *k*, it becomes increasingly difficult to prepare a representative composite. Therefore, this guidance recommends minimizing *k* in most circumstances to between 3 and 6, with a maximum of 10.

  - Site quality control results that demonstrate that the error in composite formation is minimized. This may be demonstrated via a matrix spike to one of the increments forming a composite. The analytical result for the matrix spike composite sample should approximate 1/*k* times the matrix spike concentration.

  - The ratio of the per sample analysis cost and the per sample collection and handling costs. Composite sampling is only cost beneficial if the analysis cost savings and other benefits outweigh the additional sample collection, sample handling, and other costs associated with composite sampling.

  Once an optimal *k* is selected, the number of composite samples (*n*) analyzed will be:

$$n = N/\text{optimized } k$$

- Retesting protocols are needed whenever a threshold parameter such as a composite sample MIL is involved. In most cases, the MIL will be a fraction (as a function of *k* as discussed in Section O.3.1.2.) of the IL for discrete sampling. A composite sample result greater than the MIL would require the licensee to investigate the result and possibly analyze the composite sample increments.

- When used during a FSS, the composite data are evaluated collectively to provide an estimate of the mean/median concentration levels in each survey unit, and individual composite results are compared to the respective MILs (and $DCGL_{EMC}$s) for the composite sample spacing and are used in the statistical tests, as appropriate. Comparison against MILs provides assurance that release criteria are met.

- Provisions should be made in the plan to maintain individual composite increments for retesting.

- When composite sampling is used to provide an increased probability of elevated area detection, the maximum elevated area size and concentration of concern should be carefully evaluated and approved before the plan is implemented. Section O.4.5 of this appendix provides additional discussion regarding considerations for elevated areas.

- A clear assessment of individual and collective composite sample results should be provided in documentation to ensure conformity with release criteria.

## O.4 Composite Sampling Designs: Survey Planning Example Evaluations

### O.4.1 Mean Concentration Estimation

A licensee may choose to use composite sampling during site investigations to assist decision-making in estimating the mean concentration. The reason(s) for doing so may be to reduce overall analytical costs, to increase site coverage, or both. The target population of interest could range anywhere from the site as a whole to a specific survey area/unit, depending upon the decommissioning phase. A licensee may consider composite sampling for mean estimation during any phase of the site decommissioning, including the FSS. In many cases, composite sampling results are expected to be compatible with the hypothesis tests recommended in MARSSIM (e.g., Sign Test using Scenario A)[4]. Simulations comparing power for composite sampling versus discrete sampling could be used to provide support for the method selected.

When an estimate of the study area mean is the sampling goal, the sample plan (e.g., ordinary random, systematic, stratified) is selected based on the expected distribution of the ROC—homogenous, heterogenous, spatially related—and the overall DQOs. The user then calculates the number of samples (*n*) necessary to estimate the mean objective, in accordance with the anticipated data use endpoint.

However, the loss of study area variability information with composite samples should be considered if composite sampling is intended for other uses besides the mean estimation. If the overall variability is underestimated, insufficient power and failure of the survey unit may result due to insufficient sampling for Scenario A.

The NRC staff should review the use of composite samples to estimate the mean concentration within a study area to ensure that the licensee has provided the information listed below.

[4] The number of samples (prior to compositing) should be determined based on agreed upon alpha and beta errors, and the relative shift (see MARSSIM, Rev. 1 for additional information). Because the relative shift is a function of variability, composite sampling results are not recommended for determining variability for the purpose of calculating the relative shift during the planning stage of the FSS due to the tendency for composite sampling to underestimate variability.

- When the composite data are only intended to estimate the mean concentration, information on the individual increments is not important. The mean is calculated using the standard expression(s). However, there are two additional conditions if an IL for an individual increment is a project requirement:
  - A MIL should be determined
  - Individual increments must be maintained for reanalysis, if necessary.
- When the data are intended for both estimating the mean and determining if a threshold could be exceeded, the reviewer should also ensure that other conditions are satisfied. These begin with determining if the MDC of the selected analyses are several factors less than the expected ROC concentrations or IL, such that the concentrations of interest are not masked by limited analytical sensitivity. An example of such a condition and assumed parameters is provided next.

#### *O.4.1.1 Composite Sampling Example 1—Evaluation of the Analytical MDC Against the MIL*

In this example, the ROC is Th-230. A discrete sample investigation level of 10 pCi/g is established. The selected analysis type is gamma spectroscopy with a 4-hour count time. The analytical MDC is 5 pCi/g. Five increments will be used to form the composite.

With the discrete sample IL of 10 pCi/g, the analytical result must be capable of identifying composite samples that exceed the MIL. The default MIL is calculated as the 10 pCi/g action level divided by the number of composite increments, or 10/5 = 2 pCi/g.

Because the analytical MDC is greater than the IL and calculated MIL, composite sampling cannot be applied without a more sensitive analysis, such as alpha spectroscopy. Alternatively, if the number of increments, *k,* is reduced to 2, the default MIL becomes 10/2 = 5 pCi/g and the MDC requirement is met.

This example may also be applicable for the other composite sampling data end uses discussed in this guidance.

### O.4.2 Identification of a ROC Presence/Absence Trait

Early in the site decommissioning process, composite sampling may be beneficial to assist the user in determining the proportion of the site or another population subset that exhibits a particular trait. Two specific examples include areal ROC differentiation across the site and area classification. The resultant data may be used both to provide an estimate of the proportion of the population that exhibits a specific trait and when additional testing is necessary.

(1) This approach could be used to establish what proportion of the site could be considered for varying classifications. A simple example is determining the approximate proportion of a site that is affected above the analytical detection limit for a ROC that is not present in background.

(2) Identification of which ROCs are affecting site populations when the specific ROCs affecting a site are expected to vary over distinct site regions.

With the objective being a presence/absence trait, a binomial distribution may be applied, and the number of positive composite results can be used to estimate the proportion of the site with the trait of interest. The estimated proportion ($p$) of the population with the trait is a function of the positive results ($x$) among $m$ composite random samples. $p$ is $\approx x/m$. The number of positive results is dependent upon k. This interrelationship may therefore be expressed as:

$$p = 1 - (1 - x/m)^{1/k} \text{ (EPA, 2002)}$$

This approach may be further expanded to assist in the data gathering required to assess the presence or absence trait of elevated areas of HTD radionuclides within a specified unit area. The following considerations should be addressed.

(1) The *a priori* elevated area size and concentration of concern based on a dose consequence evaluation.

(2) The probability of not hitting an elevated area of the determined size; defined as 1-β, the acceptable risk is β.

(3) The probability of the existence of elevated areas based on sampling results.

These are the sample planning considerations. Once the plan is developed, there will be further evaluations and considerations that should include statistical inferences relative to probabilistic determination of the elevated area maximum size that could be missed by the plan design. Regulatory concurrence of unidentified elevated areas of concern would therefore be necessary. Once data are evaluated, further determinations can be made as to the probability of any elevated area being in the survey area and statistical inferences determined as to the maximum probable level. A suggested approach for bounding this maximum probable concentration is to apply Chebyshev's inequality. Chebyshev's inequality guarantees that no more than $1/k^2$ of the distribution's values can depart more than k standard deviation from the mean (in this instance, k = any real number >0 rather than, as previously defined, the number of increments). From the maximum level and size inference, the dose consequence may be calculated and further assessed.

### O.4.3 Composite Sampling to Augment Required $MDC_{SCAN}$ and/or Analytical Cost Reduction during Final Status Surveys

A site may choose composite sampling as a means to resolve an FSS scenario where satisfying the required $MDC_{SCAN}$ for the applicable $DCGL_{EMC}$ results in an inordinate number of samples and an associated increase in sample analytical costs, which can be prohibitive. Other decommissioning survey phases may also select this approach to reduce the analytical costs.

Composite sampling can be used to satisfy the sample spacing requirements to detect elevated areas of concern in Class 1 FSS units for cases where it proves difficult to achieve the required $MDC_{SCAN}$ conditions. The sample plan should provide sufficient detail to demonstrate that full consideration has been given to conditions that could deleteriously affect the decision-making process.

### *O.4.3.1 Composite Sampling Example 2—Evaluation of $MDC_{SCAN}$ Requirements and Adjustments to the Number of Samples*

An example of the approach is presented here. The ROC is Pu-238. The $DCGL_W$ for Pu-238 based on dose modeling is 31 pCi/g. The gamma $MDC_{SCAN}$ (FIDLER NaI detector) is 740 pCi/g. A survey unit area size of 2000 $m^2$ is selected. The Sign test has been selected for the statistical test. For this problem, the example $DCLG_{EMC}$s for Pu-238 based on dose modeling are found in Table O.2.

**Table O.2 $DCGL_{EMC}$ for Pu-238**

| Pu-238 $DCGL_{EMC}$ (pCi/g) | | | | |
|---|---|---|---|---|
| 10 $m^2$ | 20 $m^2$ | 50 $m^2$ | 100 $m^2$ | 200 $m^2$ |
| 3,249 | 1,674 | 685 | 347 | 174 |

For this FSS example, assume a site's DQO planning inputs result in 15 required samples. The maximum size for a Class 1 survey unit is 2000 $m^2$. The respective area represented by each sample is therefore 2000/15 or 133 $m^2$. The $DCGL_{EMC}$ for a 133 $m^2$ area is determined based on a fit of the data to an area factor curve and calculated to be 260 pCi/g. The actual $MDC_{SCAN}$ (740 pCi/g) is compared with this required scan MDC (260 pCi/g). Because the actual $MDC_{SCAN}$ exceeds the required $MDC_{SCAN}$ activity concentration level, sample spacing must be reduced to account for potential elevated areas. Based on the fit of the data, a $DCGL_{EMC}$ of 740 pCi/g would be acceptable for a sample spacing of 45 $m^2$. Therefore, 44 samples would be necessary to ensure the $MDC_{SCAN}$ is adequate to detect elevated areas of the corresponding magnitude.

The site requests that sample sizes remain essentially equivalent to the original plan, due to the costs associated with analyzing 29 additional samples in multiple Class 1 survey units. Therefore, composite sampling and retesting are proposed and factored into the FSS plan to minimize the additional analytical costs. To do so, the DQOs must be planned accordingly and included in the decision-making process. This example is further developed below as an illustration.

Figure O.1 (left) shows the initial 15-sample design. However, elevated area or hot spot considerations require that the sample size be increased to 44. The original 15-sample design is maintained by forming composite samples composed of 2 to 3 increments each, as represented in Figure O.1 (right). The revised design sample spacing is 45.5 $m^2$; the 3-increment composite therefore represents a sample area of approximately 136.4 $m^2$. This design will provide the necessary assurance that either sampling or scanning will identify any elevated areas of concern. Each of the 44 increments is collected, composites are formed, individual increments are maintained for retesting as needed, and the composites are analyzed. The composite results are then compared with the appropriate MIL. A composite result less than the MIL provides the evidence that elevated areas are below the action threshold and composite results greater than the MIL will require retesting of the increments before a final determination is made as to whether further investigations are required. The example assumes that any elevated areas smaller than 45.5 $m^2$ with activity levels greater than 740 pCi/g would be identified during the scanning phase of the survey. If increment retesting or surface scanning support the need for follow-up investigations, additional measurements will help determine the extent and magnitude of elevations above the DCGLw and to assess whether the elevated areas would cause the release criteria to be exceeded.

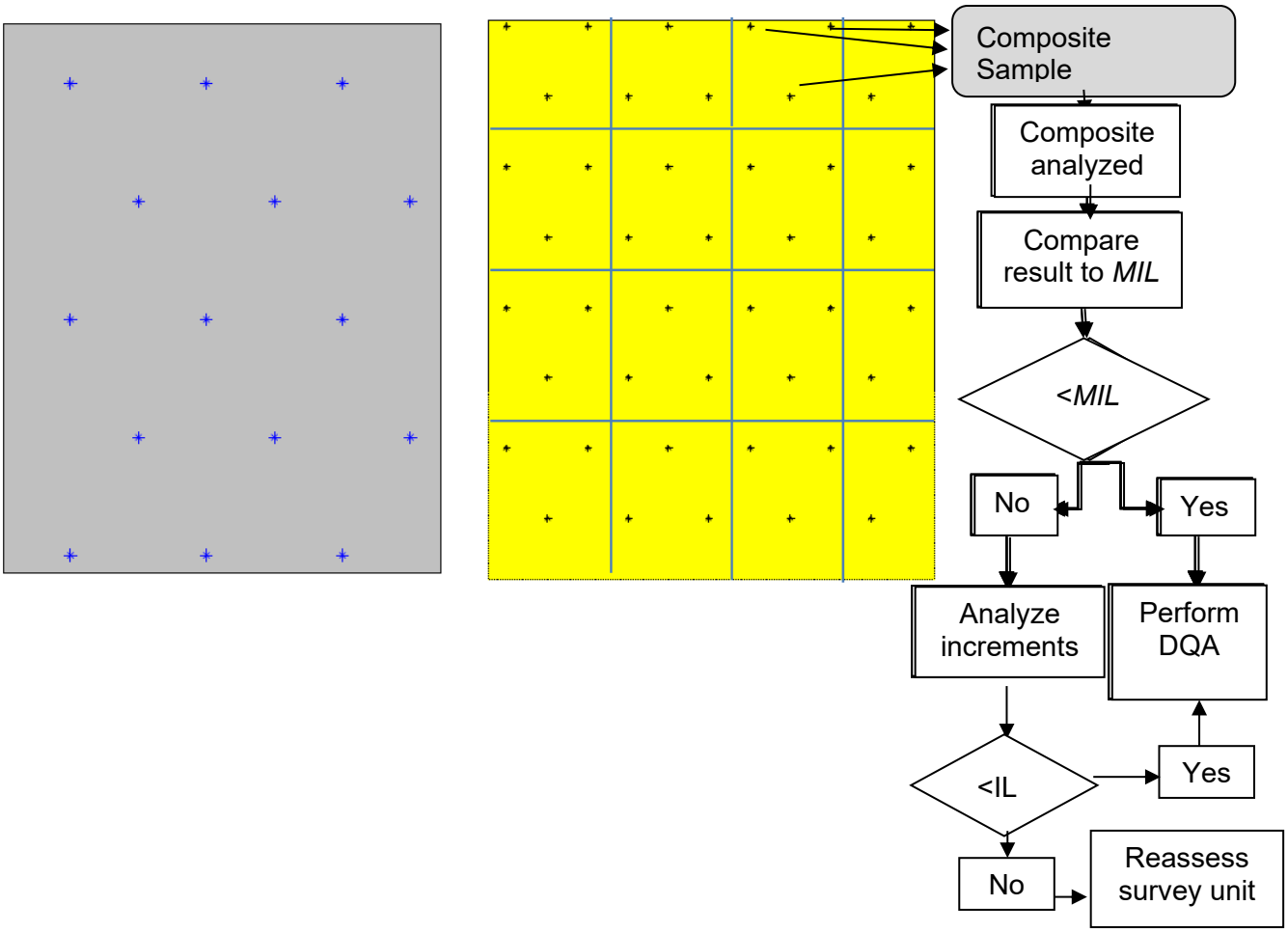


**Figure O.1 Pu-238 Sample Number Comparison** *(the Left Figure Shows a Pu-238 Survey Unit with 15 Sample Locations; the Right Figure Shows a Pu-238 Survey Unit Showing 44 Sample Increment Locations and 15 Composite Groupings)*

The DQA phase will need added rigor to close the differential between the required $MDC_{SCAN}$ that was addressed by increasing the sampled locations to 44 and forming the 15 composites. The data assessment would compare each composite concentration result with an appropriate MIL that accounts for the various scenarios that could exist. There are multiple combinations of increments adding elevated concentrations of the ROC to the composite results. For this example, the scenarios to consider for an MIL exceedance could be as follows:

(1) one increment adding a high concentration

(2) two increments adding moderate-to-high concentrations

(3) all three increments adding low-to-high concentrations

These assessments will lead to varying decisions, including no further investigation necessary, failure of the Elevated Measurement Comparison, or retesting increments and comparing the results to the applicable $DCGL_{EMC}$. Table O.3 illustrates three scenarios that are based on the number of contaminated (involved) increments for a composite sample. The scenarios show the MILs that are a function of the area represented by the number of involved composite increments with Pu-238 concentrations that equal the respective $DCGL_{EMC}$. The results show three differing concentration values where the potential exists for exceeding a $DCGL_{EMC}$. Because of these various scenarios, the composite sample MIL must be established at the lowest concentration value—or 244 pCi/g in the Table O.3 example. Any composite result greater than 244 pCi/g would need retesting of the individual increments that comprised the suspect composite sample.

The conclusion of the example is that, for the established parameters, using the composite sampling approach will provide a high degree of certainty that both the average residual Pu-238 can be readily determined and elevated areas of concern will be identified with minimal false negatives when composite sampling with a defensible MIL is combined with surface scanning. Lastly, the analytical cost for the survey was maintained, although there will be additional field labor costs to collect, package, and record the composite sample and the increments.

**Table O.3 Composite Result Investigation Level Evaluation**

| Area ($m^2$) | $DCGL_{EMC}$ (pCi/g) | Involved Increments/ % Activity Weighted Contribution at the $DCGL_{EMC}$ | | MIL (pCi/g) |
|---|---|---|---|---|
| | | Number of Increments | % Activity Contribution | |
| 136.5 | 254 | 3 of 3 | 1 | 254 |
| 91 | 378 | 2 of 3 | 0.67 | 253 |
| 45.5 | 740 | 1 of 3 | 0.33 | 244 |

% assumes other increment(s) have no added activity.

### O.4.4 The Contaminant $DCGL_W$, Mean, Variability, and/or Decision Error Combinations Result in a Relative Shift <1

MARSSIM recommends that the relative shift be maintained between a value of 1 and 3, such that a reasonable sample size results during the FSS. Site-specific conditions could lead to situations where the relative shift is less than one, requiring a large number of samples (N) to achieve adequate statistical power. Such conditions, either individually or in combinations, could include a high expected mean concentration (LBGR) relative to the $DCGL_W$, a high degree of variability within the survey unit, or both. Current MARSSIM guidance recommends the following when this situation occurs: either reducing the value of the LBGR, which affects the concentrations at which a Type II error could occur or revisiting the DQOs together with the regulatory authority.

The basic two options available to the site for revising the DQOs such that sample sizes are reduced—other than reducing the LBGR—are increases in either the $DCGL_W$ or the Type I error. Both options require regulatory interactions and approvals. Another option that a site may consider is composite sampling, where the initial MARSSIM-designed sample size forms

the basis for the number of increments (k). The site would therefore need to determine the number of composite samples that will be a function of an optimal k value. Section O.3.2 contains the considerations for k that the licensee must evaluate.

### O.4.5 Classifying Survey Areas as Containing Elevated Areas

Section O.4.2 introduced the increased probability for elevated area detection, although when not performed correctly, the opposite effect could occur, where an elevated area is masked. The advantage for elevated area detection using composite samples is that, through the reduced analytical costs that can be achieved, more of the budget is available for sampling. This leads directly to better areal coverage and hence increases the probability that a sample location will fall on an elevated area. Therefore, a composite sample approach can provide greater confidence for detecting elevated areas of those ROCs considered to be HTD radionuclides (e.g., Sr/Yttrium-90, C-14), when a surrogate is not available. Therefore, an affordable FSS plan can only be developed by first deciding upon an acceptable elevated area size and concentration magnitude.

Again, this guidance provides an approach to overcome the limitations of MARSSIM in cases where HTD radionuclides are present in elevated areas. The guidance presents an approach that can more easily meet demanding sample requirements for statistical tests and for addressing elevated areas when HTD radionuclides are present. For this guidance, such a technical approach might include composite sampling, composite sampling further supplemented with rank set sampling, or the adaptation of a probability-based design for locating elevated areas of a predetermined size and shape (EPA, 2002; G.P. Patil, "Ranked Set Sampling," 2002 (Patil, 2002b); J.E. Jozani, "Design Based Estimation for Ranked Set Sampling in Finite Populations. Environmental and Ecological Statistics," 2010 (Jozani and Johnson, 2010; R.O. Gilbert, "Statistical Methods for Environmental Pollution Monitoring," 1987 (Gilbert, 1987)). However, before preparing either design, an *a priori* elevated area size of concern and the associated $DCGL_{EMC}$ must be determined. Once the elevated area size is determined, then either a discrete or composite sampling approach can be applied to provide a high level of probability that the elevated area will be sampled.

There remains yet another problem associated with this *a priori* elevated area. The problem is how to select the *a priori* size, as there will likely remain stakeholder concerns for smaller elevated areas that could again be missed. Therefore, the technical justification should include additional dose modeling details as to the potential impacts of smaller elevated areas that could go undetected and the potential contribution to the total dose from all remaining source terms across the site. Other approaches that may be considered include maximum concentration bounding scenarios using Chebyshev's inequality or a Bayesian approach to estimate the maximum potential contamination level at the site.

In summary, the HTD elevated area identification condition will need significant evaluation and discussions with appropriate survey designers and regulatory authorities.

## O.5 Conclusion

In a "typical" radiological FSS, the reduced probability of identifying elevated areas is accounted for through the iteration of adjusting sample spacing to satisfy required scan MDC sensitivity. To limit the increase in sample sizes, more licensees are suggesting composite sampling as one method to control the increased analytical costs that result. As with most sampling

approaches, there are both advantages and disadvantages that require evaluation and review to ensure that the plan accounts for any limitations.

With this information, robust radiological sampling plans can be developed that will address multiple issues while providing assurance on decisions about the average residual concentrations across the survey unit as well as elevated area considerations. The number of composite samples and composite increments can be controlled, based on the type of contaminant, $DCGL_W$ levels, analytical MDCs, and the corresponding appropriate MILs. Additional guidance on optimizing all factors may be found in the references.

## O.6 References


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# APPENDIX P

# FRAMEWORK FOR USE OF ENGINEERED BARRIERS AT DECOMMISSIONING SITES

# P.1 Framework for Use of Engineered Barriers at Decommissioning Sites

The purpose of this appendix is to provide guidance to licensees considering the use of engineered barriers to meet radiological criteria for license termination (e.g., engineered covers, including erosion protection designs, stabilizing cementitious materials, and reactive walls). Guidance topics include use of a risk-informed, graded approach for selection of engineered barriers, the engineered barrier analysis process, support for engineered barrier performance, and the need to consider potential engineered barrier degradation mechanisms. A detailed example of the application of a graded approach to the design and implementation of engineered barriers for erosion protection is also provided.

The guidance is directed towards licensees pursuing restricted use, since it is envisioned that this is the situation where engineered barriers will most frequently be used. The guidance is directed towards designing new engineered barriers; however, it is recognized that some sites may have *in situ* engineered features that are part of the existing site being decommissioned, for which the licensee wants to take account. In these situations, elements of P.1.2 and P.1.3 may be most helpful. The guidance is intended to strike a balance between providing adequate direction and being overly prescriptive. Because the application of engineered barriers is site-specific (e.g., due to different radiological source terms, different exposure environments, and different natural systems), some elements of the guidance may not be applicable to every site.

## P.1.1 Risk-Informed Graded Approach for Engineered Barriers

The general need for an engineered barrier at a specific decommissioning site should be determined by considering if the barrier would be needed for compliance with the LTR dose criteria (e.g., mitigates the impact of natural processes such as erosion and infiltration). Once the general need is determined, a risk-informed graded approach should be used for selecting and designing engineered barriers at specific sites and documenting the technical basis for the selected barrier. The risk-informed, graded approach to engineered barriers is similar to the risk-informed graded approach for institutional controls described in Section M.2 of Appendix M of Volume 1, which consists of a general risk framework and associated grades of institutional controls. The same risk framework is used for both institutional controls and engineered barriers and is defined by the hazard level and likelihood of hazard occurrence[1]. The NRC's philosophy is that robust engineered barriers and additional bases for engineered barrier performance should be provided for higher risk sites (compared to lower risk sites). As described in Section M.2, higher risk sites are those sites with projected doses greater than 1.0 mSv/y (100 mrem/y) with no institutional controls in place and sites that must rely on engineered barriers for relatively long periods of time (i.e., longer than 100 years) due to the presence of risk-significant quantities of long-lived radionuclides. Uncertainty in the performance of engineered barriers increases with time with limited data available on long-term performance. This coupled with an increasing likelihood of disruption of the engineered barriers makes it especially important to design a robust cover. Generally, "robust" means more substantial, reliable, and sustainable for the time period needed without reliance on ongoing active maintenance. The term "robust" is similar to the term "durable" used for institutional controls at higher risk sites (see Section M.2 of Appendix M of Volume 1). Section P.2 provides a detailed example of the application of the risk-informed graded approach to engineered

[1] It is important to note that hazard is different from risk. Hazard is primarily based on the inventory and dose conversion factors, while risk also considers the ability of the radioactivity to be released. Therefore, a site can have a high hazard but low risk, if the radioactivity is sufficiently contained or access to the radioactivity is significantly limited. Engineered barriers can effectively reduce risk for otherwise high-hazard sites.

barriers for erosion protection. Robust engineered barriers are also needed for sites where the hazards are being significantly mitigated by the functioning of a barrier (i.e., risk reduction). "Significantly mitigated" is site-specific and can vary due to differences in the source and other features of the site (see example). In general, an engineered barrier that reduces the risk by more than a factor of 5 would likely be considered to be significant. The basis for this factor is that, in many circumstances, the uncertainty in the risk for restricted use sites can approach or exceed this value.

**(Example)** Site A and Site B both contain soil contaminated with equal concentrations of Sr-90. The primary exposure pathway is from leaching of the contamination to groundwater. Both sites design an engineered cover to limit infiltration for the next 100 years. Site A has a thick unsaturated zone composed of a clayey soil. The estimated travel time to the groundwater, constrained by observations of past releases of Sr-90, is at least 200 years. Site B has a thin unsaturated zone composed of a sandy soil and the estimated travel time to the groundwater is 60 years. There are no additional data to constrain the numerical estimate of the travel time at Site B.

**Conclusion:** Site B would need a greater technical basis for the performance of its engineered cover compared to Site A, because Site B would have less risk reduction from the natural system and the travel time estimate (which is directly tied to risk via radioactive decay) is more uncertain.

Although more frequently used at higher risk restricted use sites, robust engineered barriers may also be used at lower risk restricted use sites to limit reliance on active monitoring and maintenance. Such an approach might alleviate institutional controls and maintenance and thereby reduce licensee costs. In some limited cases, a robust engineered barrier could also be used at an unrestricted use site, because it has been designed so that its performance does not rely on active ongoing maintenance.

The concept of passive performance of an engineered system is not unique to decommissioning of a radiologically contaminated site. There are many long-lived man-made structures that have not benefitted from continual monitoring and maintenance. However, passive performance of an engineered system cannot be assumed. As indicated in the sections that follow, passive performance must be justified through design, experimentation, analysis, and support, and uncertainty must be considered in projecting performance, particularly over periods of time that exceed the experience base. In fact, the experience base is oftentimes much less than the time period relied on for performance for most waste disposals.

Uncertainty related to engineered barrier performance over a long temporal scale is also contextual in that the length of experience in the use of that type of barrier should be considered relative to how long lived the contamination is. In general, hundreds to thousands of years would be considered a long time when applying engineered barriers to decommissioning, because it is almost certain that site-specific experience for the performance of an engineered system does not extend beyond several decades. Section P.1.2 provides guidance on how the barrier analysis process should be used to determine the performance of the engineered barriers. The risk-informed graded approach to engineered barriers is linked to the sections on effective barrier analysis (Section P.1.2) and the technical basis for engineered barrier performance (Section P.1.3), as well as to Section 17.7.4 of Volume 1, on maintenance and monitoring. For example, where there is a demonstration that the engineered barriers have

been designed to be robust and not reliant on ongoing active maintenance and repair, the amount of information and resources needed to support maintenance would be considerably reduced.

| |
|---|
| The robustness of an engineered barrier and the amount of technical basis provided for an engineered barrier should be commensurate with the level of risk for a site (i.e., lower for lower risk sites or higher for higher risk sites) and the amount of risk reduction assumed for the barrier.<br><br>More robust barriers or strong support for engineered barrier performance is needed for higher risk sites or where hazards at a site are significantly mitigated by the barrier. |

*Engineered barriers should be designed with the goal of remaining effective over the time period* needed to achieve compliance (as long as necessary and practical during the 1000-year compliance period at a minimum), especially for long-lived radionuclides. The following items should be considered in designing engineered barriers for decommissioning sites:

- Designs should simplify long-term control and minimize the reliance on active ongoing maintenance and associated costs, especially for sites with long-lived radionuclides.

- Designs for sites with long-lived radionuclides (hundred years or more) should place more reliance on natural materials and less reliance on synthetic materials that are less proven over the long term (e.g., less reliance on compacted clay and more reliance on enhanced evapotranspiration). For sites with short-lived radionuclides (tens of years or shorter), synthetic materials may be more advantageous, because there is less variability in their performance.

- The determination of adequate financial assurance should consider the cost of monitoring and routine maintenance and the need for potential major repairs of the engineered barrier, based on the time period during which the engineered barrier performance is relied on to achieve compliance.

### P.1.2 Engineered Barrier Analysis Process

This section is directed towards licensees pursuing restricted use. Licensees using engineered barriers for unrestricted use may find parts of this section useful but should only credit the passive performance (without monitoring, maintenance, and inspection) for mitigating radiological impacts.

Decisions on approving the use of engineered barriers to meet the LTR will be performance-based using the dose criteria of the LTR. Therefore, licensees have flexibility in how engineered barriers may be used together with other remediation activities to achieve the dose criteria. In addition, staff will base the approval decisions using a risk-informed approach. In their proposal to use engineered barriers, licensees should include all relevant information concerning the risks of using the approach versus other remediation alternatives.

To implement a risk-informed graded approach, licensees should accurately assess the performance of the engineered barriers. Using engineered barriers for compliance with Subpart E for restricted use, entails an analysis of the following two items under 10 CFR 20.1403(b) and (e), respectively:

- the effectiveness of the engineered barrier in reducing risk when institutional controls are in place (i.e., active monitoring and maintenance are in place to help ensure a properly functioning cover)

- the effectiveness of the engineered barrier in reducing risk under a loss of institutional controls (including loss of active monitoring and maintenance) that may lead to more rapid degradation of engineered barrier performance compared to the case where institutional controls are in place (given uncertainty in long-term performance of engineered barriers, degradation should be studied through sensitivity analysis).

The first analysis might evaluate the dose to the average member of the critical group who is assumed to be located at the site boundary due to active institutional controls that prevent access to the site. The performance of the engineered barriers could include the benefit of active monitoring and maintenance that might increase the level or length of performance assumed for the engineered barrier relative to the case with no monitoring and maintenance. The dose limit specified in 10 CFR 20.1403(b) when institutional controls are in place is 0.25 mSv/y (25 mrem/y).

The second analysis, which assumes a loss of institutional controls, would typically include evaluating exposure scenarios involving human receptors occupying or otherwise participating in onsite activities immediately following license termination or at time=0 years. The dose limit applied to an average member of the critical group under a loss of institutional controls is 1.0 mSv/y (100 mrem/y) (or 5.0 mSv/y (500 mrem/y), if certain additional criteria specified in 10 CFR 20.1403(e)(2) are met). The analysis should consider the impact of disruption of the engineered barriers from human activities (e.g., well drilling, home construction, resource exploitation), and other natural or human-induced degradation mechanisms, events, or processes (e.g., root penetration into an engineered cover [see for example DOE, 2009] or infilling of a lateral drainage layer) under a loss of institutional controls. In some cases, a receptor may disrupt the engineered system that could result in higher doses through a different pathway (e.g., an agricultural intruder disrupts an engineered cover with plowing but does not disrupt the contamination directly). Disruption of the cover could result in increased infiltration and higher groundwater pathway doses.

The analysis should also consider reasonably foreseeable and less likely but plausible events and processes that could lead to disruption of the engineered barriers. “Reasonably foreseeable” disruptive conditions from humans or natural events and processes are those processes and events expected to have a probable or likely occurrence over the analysis period. “Less likely but plausible” events and processes are less likely than reasonably foreseeable events and processes but should also be considered to better understand challenges to the system, evaluate uncertainty in engineered barrier performance, and provide additional risk information to the decision-maker. A features, events, and processes (FEPs) evaluation can be a useful tool for determining what FEPs should be considered. The intent of performance assessments is to provide reasonable assurance, considering uncertainties in engineered and natural systems over long time periods that the actual performance of the disposal facility will comport with the modeled or assumed performance. Natural or human processes and events are possible stressors to the engineered barrier design. On the other hand, highly unlikely or implausible processes or events would not be expected to have a significant influence on risk because of the low probability of occurrence and, therefore, need not be evaluated.

The potential impact to engineered barriers from reasonably foreseeable disruptive conditions from humans or natural events and processes can vary by site and by barrier. In some circumstances, it may be reasonable to take no credit for the engineered barriers. For example, a hypothetical site installs a geomembrane at the surface to control infiltration at a restricted use site. In analyzing a loss of institutional controls, it would be reasonable to assume no performance credit for the geomembrane, because it could be easily removed or damaged by near surface processes, and this type of barrier is subject to discrete failures. On the other hand, consider a site in a rural setting using a large, thick earthen cover of low gradation. Complete removal (and complete degradation) of the earthen cover under these conditions, while possible, may not be considered reasonably foreseeable.

Realistic exposure scenarios for future site use for this example (e.g., residential or farming) would dictate the reasonably foreseeable disruptive conditions from humans would commonly be construction of a foundation and a well for a residence that disrupts a portion of the cover, or farming of the site, which would result in plowing the top layers of the cover and the resultant potential impacts from the rooting of vegetation and increased infiltration. Because it may be difficult to (1) estimate the likelihood of disruption of engineered barriers or (2) predict the long-term performance of engineered barriers, the decision maker should obtain additional risk information by evaluating less likely but plausible events that could challenge the system, although it is not necessary to strictly compare the results of these less likely but plausible events and processes to radiological criteria for license termination.

With regard to engineered barrier degradation, the licensee should address the two cases where maintenance is either in place or lost, because the assumption for loss of institutional controls includes the loss of maintenance of engineered barriers and physical controls such as fences or signs. It should be noted, however, that, for those cases where an erosion control cover is designed in accordance with the uranium mill tailings guidance in NUREG-1623, “Design of Erosion Protection for Long-Term Stabilization,” issued September 2002, a case might be made for a durable, long-lived engineered barrier that does not rely upon ongoing active maintenance (i.e., maintenance needed to ensure that the design will meet specified longevity requirements) and associated future costs to maintain the erosional stability of the site. For this case, a licensee may be able to demonstrate that no significant degradation of the erosion control cover is expected.

The licensee should use the barrier analysis process to determine how much credit is being taken for engineered barrier performance in reducing risk. To accomplish this, the licensee should perform an analysis with the engineered system present and functioning. However, for unrestricted use or when evaluating the loss of institutional controls (without active monitoring and maintenance), the licensee should assume the performance of the barriers may degrade over time. An assumption of instantaneous and complete failure of a barrier is not required. The goal is to clearly identify the expected benefit of the engineered barriers quantitatively in terms of dose reduction. The analysis should also address the uncertainty in the expected performance of the barriers. For example, a comparison could be made of the doses without engineered barriers present (all off) (or with degraded engineered barriers) to the doses with engineered barriers performing as designed and expected (all on). The assumption to model a degraded barrier is generally more realistic than the assumption of the absence of a barrier and actually may lead to higher doses in some situations; for example, partial failure of a cover or partial failure of a grout wall system can focus water flow and create a “bathtub” effect. Thus, the licensee should use caution when making simple on-off analyses (i.e., where the barrier is either assumed to be completely present or completely absent). For some barriers an

appropriate approach to defining a degraded condition or an end-state of ecological succession may be to use information from analog sites.

Caution should also be used when analyzing systems with multiple engineered barriers. When multiple barriers are present, the performance of one barrier may mask the potential contribution of another barrier to reducing risk (e.g., when redundant barriers are present). In these situations, the analysis may need to evaluate various combinations of barrier performance to determine the individual and cumulative contributions (positive or negative). In some cases, characteristics of one barrier may challenge or impair the performance of another barrier. These cases should be carefully assessed (e.g., increased infiltration resulting from eliminating vegetation by inert covers or the clogging of a drainage layer with a resultant hydraulic head over the waste). The licensee should determine the type of analysis on a case-by-case basis. For the complex and higher risk decommissioning sites and those sites with long-lived radionuclides, the use of a probabilistic analysis should be strongly considered, because a deterministic analysis may not be able to adequately address the uncertainty in the calculations. However, for simpler, lower risk sites and sites with short-lived radionuclides, a deterministic analysis with a sensitivity analysis may be sufficient. Deterministic methods should be used to select the design flood for the development of long-term erosion controls (see Appendix P).

In summary, the analysis of engineered barriers should identify and evaluate those conditions or processes that are adverse to performance and result in noncompliance. The following should be provided:

- engineered barrier analysis with institutional controls in place (i.e., taking credit for active monitoring and maintenance) for restricted use sites

- engineered barrier analysis assuming a partial or total loss of institutional controls (i.e., not taking credit for active monitoring and maintenance) for unrestricted and restricted use sites

- engineered barrier analysis considering reasonably foreseeable and less likely but plausible disruptive human and natural events and processes that may lead to degradation of the engineered barriers for unrestricted and restricted use

### P.1.3 Technical Basis for Engineered Barrier Performance

Significant uncertainty exists concerning predicting the service life and long-term degradation rates of most engineered barriers. This section provides guidance on the main elements necessary to support the analysis and evaluation of engineered barrier performance in Section P.1.2, including the following:

- the design, features, and functionality of the engineered barriers

- the technical basis for assuming the barriers will meet the dose criteria, including information about the length of time relied on for performance given the radionuclide mix present and considering the degradation mechanisms and the combined and synergistic effects resulting from the real-world conditions expected for the barriers over the performance lifetime (Section P.1.4)

- the uncertainty in parameters and models used in assessing barrier performance and the design of engineered barriers

- the suitability of numerical models used for validating engineered barrier performance
- a parametric or component sensitivity analysis to identify how much degradation of the engineered barrier would result in noncompliance (see Section P.1.2)
- model support for engineered barrier performance (e.g., analogs, experiments, simple engineering calculations to demonstrate reasonableness of the results)
- quality assurance and quality control (QA/QC) for the design and analysis of engineered barriers

> Analysis can be used to understand the impact of uncertainty. For example, 1/4 loss of a cover may not result in non-compliance but loss of 3/4 of a cover would. Then determine if reasonably foreseeable (or less likely but plausible) natural and human processes might cause a loss of 3/4 of a cover. This analysis approach may be an additional way to deal with the uncertainty due to lack of data for long-term cover degradation.

Many engineered barriers are not amenable to model validation in the true sense; therefore, multiple lines of evidence are recommended. Model support can come in many different forms, including but not limited to: analogs, laboratory experiments, field experiments, formal and informal expert judgment, and engineering calculations to demonstrate the reasonableness of the results (e.g., hand calculations when numerical models are used). The level of model support should be commensurate with the risk significance of the engineered barriers to achieving site decommissioning (see Section P.1.1). If the level of performance needed for an engineered barrier is consistent with past experience at similar sites, and the engineered barriers have similar designs and QA, then the model support could be considerably less than for an engineered barrier with performance objectives that significantly exceed engineering experience. When considering engineering experience, the licensee must take care to ensure that the environmental conditions for the relevant degradation mechanisms are reasonably similar, since many degradation mechanisms can be very sensitive to environmental exposure conditions.

For engineered barriers that require very long-term performance, natural analogs should be considered to help define long-term conditions of the barrier and its environment. The greatest uncertainties stem from extrapolating the results of short-term tests and observations to long-term performance. Standard engineering approaches frequently implicitly assume that the initial conditions persist, while the actual barrier more appropriately could be viewed as an evolving component of a larger, dynamic ecosystem (Waugh, “Ecology, Design, and Long-Term Performance of Surface Barriers: Applications at a Uranium Mill Tailings Site,” 1997). In fact, NUREG/CR-7028, “Effectiveness of Engineered Covers: From Modeling to Performance Monitoring,” issued December 2011, presents the results of an engineered cover study and concludes that compacted soil materials used in cover materials at the sites studied did not retain “as built” properties over periods of regulatory interest. The properties (specifically saturated hydraulic conductivity) of these materials change to values more typical of surrounding soils within 5 to 10 years after installation.

For some types of engineered barriers, natural analogs might provide information about possible long-term changes to an engineered system and can be thought of as a long-term, uncontrolled experiment. Some parameters (e.g., permeability, density, and diffusion coefficients) of natural analogs can be used to help define anticipated end-states of some

simple engineered barriers. Evidence from natural analogs can help demonstrate that there are real-world complements to the postulated numerical predictions. It is important that the functionality of an engineered barrier be considered when developing analogs. The example of earthen mounds constructed by Native Americans, given below, presents a reasonable analog for the physical stability of an engineered cover. It does not provide analog information for the ability of engineered covers to limit infiltration or the release of radionuclides. Natural analogs can demonstrate long-term ecological conditions that may impact barriers such as plant species and spacing. Analog information is uncertain for a variety of reasons, such as unknown past environmental conditions. Therefore, analog information should not be envisioned as proof of future engineered system performance but rather as providing confidence that the engineered system is likely to perform as designed or to its final degraded end-state.

Experience is limited for the long-term performance of *most* engineered barriers. However, a number of analogs to engineered covers have shown sustained durability, even for thousands of years (discussed in more detail below). In addition, some cementitious materials used by the Romans, for example, are intact after more than a thousand years of exposure to the environment. In the United States, cements in the Erie Canal and used in colonial settlements are 200–300 years old.

When evaluating analogs, it is important to note that the structures that have persisted are most likely the most durable structures. That is why it is important to consider analogs that have persisted, as well as those that may have experienced damage or failure. An additional complicating factor is that the initial conditions and past exposure environment for the analogs are not known and may only be estimated. However, developing an understanding of analogs increases the likelihood that a design may be implemented with sustainable long-term performance. It should be reiterated that natural analogs should be only one element of the technical basis for the long-term performance of engineered barriers.

Monitoring and maintenance are needed to verify the effectiveness, durability, and service life of the engineered barriers. This monitoring involves both the environmental system surrounding the engineered facility that could be disruptive and the facility itself. (Section 17.7.4 of Volume 1 discusses the risk-informed approach to monitoring.) Nondestructive monitoring technologies that included designed and emplaced sensors are preferred to conventional post-failure monitoring. Novel ideas, such as introducing special dyes or tracers within the engineered system, may facilitate identification of impending failures. The identification of these and other measurable performance indicators within a monitoring strategy, coupled with any needed repair or remediation, can be very important in extending the effective service life of these facilities and reducing risk. To the extent practicable, engineered barriers should be designed to support and simplify monitoring and maintenance. NUREG/CR-7169, “Sensors and Monitoring to Assess Grout and Vault Behavior for Performance Assessment,” issued June 2014, presents a preliminary evaluation conducted by the National Institute of Standards and Technology on the state-of-the-art sensors, nondestructive evaluation methods, and geophysical techniques used to quantify changes to chemical (e.g., redox state) and structural properties of large engineered systems used for waste isolation. While primarily applicable to waste disposal facilities, the report provides useful information on techniques that could be used to monitor engineered cementitious material performance at complex decommissioning sites. The study focuses on measurement techniques used to detect the onset of cracks and geochemical changes. Some techniques described in NUREG-2151, “Early Leak Detection External to Structures at Nuclear Power Plants”, issued April 2013, may also be appropriate non-destructive monitoring techniques to assess the performance of barriers, especially for changes in moisture content.

**Example of Natural Analog Model Support—Native American Mounds**

With respect to the durability of earthen covers but not with respect to limiting infiltration or waste releases in the groundwater pathway, an example of a natural analog may be the various earthen mounds constructed by Native Americans that have survived for periods of 1,000–5,000 years. Archaeologists have dated the mounds by excavating bones and artifacts from the mounds and determining the age of the object or the date of its burial. Information on the mounds is readily available by visiting State or national parks associated with the mounds. Also, there is considerable information and reference material available on the Internet. Examples of Native American mounds that have survived relatively intact for very long periods of time are shown in Table P.1.

**Table P.1 Examples of Surviving Native American Mounds**

| Mound | Location | Approximate Age (years) |
|---|---|---|
| Grave Creek Mounds | Moundsville, WV | 2,500 |
| Hopewell Culture Mounds | Chillicothe, OH | 2,000 |
| Cahokia Mounds | Collinsville, IL | 1,000 |
| Poverty Point Mounds | Epps, LA | 3,500 |
| Watson Break Mounds | Monroe, LA | 5,500 |

The mounds vary in size, with some being more than 23 meters (m) (76 feet (ft)) high and more than 100 m (300 ft) long. Mounds of this size reasonably approximate the size of engineered covers that may be installed at decommissioning sites. Therefore, the long-term stability of these analog sites provides additional assurance that wastes may be effectively stabilized for very long periods. It should be noted that stability in this context refers to erosional stability and not the ability of the mounds to limit infiltration, which is unknown. More importantly, guidance for earthen cover designs could be developed by examining the causes for failures of any damaged burial mounds (where only portions of the mounds remain and can be examined effectively). Understanding the long-term performance of analog systems may allow for additional safety factors to be used in current designs.

### P.1.4 Degradation Mechanisms and Functionality of Common Engineered Barriers

The purpose of this section is to identify, for licensee consideration, the common engineered barriers, the degradation mechanisms for the common barrier types and materials, and their typical functionality. This information may help the staff or licensees to select and design appropriate engineered barriers and understand the considerations needed to assess long-term barrier degradation, to make the overall decommissioning process more efficient.

The common engineered barriers are those that may be encountered at a decommissioning site. Because technology evolves, and a site may have unique considerations, this list should

not be viewed as comprehensive. Engineered waste forms are not explicitly listed as a common barrier for decommissioning, because, in most instances, engineered waste forms such as might be present in a low-level waste disposal facility are not used. The assessment of engineered waste forms has been addressed in low-level waste disposal, and that guidance should be considered with respect to waste forms used in decommissioning (NRC's "Technical Position on Waste Form", Revision 1, January 1991). Common barriers that may be used at decommissioning sites and their primary functionality are as follows:

- Geomembranes—Geomembranes are synthetic materials used primarily to limit infiltration to the waste or residual radioactivity. High-density polyethylene (HDPE) geomembranes are increasingly used for their potential long service life and effectiveness.

- Engineered cementitious materials—Concrete/mortar/grout are typically used to stabilize waste or residual radioactivity, provide a chemically favorable environment for retention of radionuclides, limit water contact, prevent erosion, provide shielding, and reduce potential intruder contact with the waste or residual radioactivity.

- Engineered Covers—Multilayered and composite engineered covers are typically used to limit infiltration, provide for shielding between the contamination and potential receptors, eliminate exposure scenarios, and limit erosion. Covers for infiltration control are typically either a resistive type or an evapotranspiration type.

- Erosion protection barriers—Barriers or parts of a barrier intended to reduce and slow erosion. This could include a riprap layer encompassing part or all of an engineered surface cover (e.g., side slopes of covers that have relatively high gradients).

- Vertical barriers—Vertical barriers may be soil-bentonite, soil-cement-bentonite, cement-bentonite, sheet pile (steel or HDPE), and clay barriers and are primarily used to control the horizontal migration of groundwater.

- Permeable reactive wall—A contaminant plume is channeled between impervious vertical walls, referred to as the funnel, and flows naturally through a permeable reactive barrier gate, where the pollutants are treated *in situ* during the flow process.

- Interceptor trenches—Used to intercept and collect contaminant releases, they are typically only applicable with monitoring and maintenance.

- Chemical barriers—Chemical barriers are used to modify subsurface environmental conditions (e.g., pH, redox potential or Eh) to limit the solubility of radionuclides or to provide a more favorable geochemical environment for sorption. Engineered cementitious materials are a good example (see above). Because of the diversity of chemical barriers that could be applied, the degradation mechanisms and typical levels of functionality are not provided in the following sections but would need to be evaluated on a case-by-case basis.

### *P.1.4.1 Degradation Mechanisms*

The degradation mechanisms may not be comprehensive, due to the large variability in conditions and processes from site to site, but they should represent those typically encountered. Degradation mechanisms depend on both the barrier and site-specific conditions.

When evaluating degradation mechanisms, licensees should carefully consider whether the environmental conditions assumed or used in an analysis of long-term performance are representative of the *in-situ* conditions expected for the engineered barrier. The main degradation mechanisms are described below for the different engineered barriers.

*Degradation of Cement-Based Engineered Barriers*

The major environmental degradation processes that affect cement-based engineered barriers are sulfate attack, carbonation, corrosion of reinforcing steel, alkali-aggregate reactions, and leaching by acidic subsurface water. Other degradation mechanisms include freeze-thaw deterioration and microbiological attack. Center for Nuclear Waste Regulatory Analyses (CNWRA) 2009-001, "Review of Literature and Assessment of Factors Relevant to Performance of Grouted Systems for Radioactive Waste Disposal," (issued April 2009) summarizes important degradation mechanisms for cementitious materials, many of which are discussed below. Degradation mechanisms can also include poor design and construction of cement-based structures, and can include differential settlement of the structures, stress concentrations, and seismic effects, as well as insufficient structural engineering design. To avoid these latter degradation mechanisms, the structures need to be properly designed and constructed under strict QA/QC procedures to ensure that they meet their design objectives. The discussions that follow will not address the design- and construction-related degradation issues of cement-based materials.

*Sulfate Attack*

Sulfate attack is a process by which sulfate ions in solution chemically react with compounds in hydrated cement (e.g., portlandite, calcium aluminate, and calcium-silicate-hydrate) leading to expansion, strength loss, and potential disintegration of the cement. Naturally occurring or anthropogenic sources of sulfates of sodium, potassium, calcium, and magnesium are sometimes found in subsurface water and soils. The cement itself can also be a source of sulfate. Sulfate attack has occurred in several regions of the United States, particularly in arid regions such as the Northern Plains and the Southwest States. Localized sources of sulfates in groundwater include mine workings, mine tailings, blast furnace slag waste piles, and chemical waste ponds. Water used in irrigation can also be a potential source of sulfate attack, due to the gradual accumulation of sulfates in the soils.

Solutions with sodium and magnesium sulfate can react with portlandite and calcium silicate hydrate (C-S-H) and lead to the formation of the expansive phase gypsum. C-S-H is the major binding component of hydrated cement and loss of C-S-H can lead to a loss of strength and disintegration of the hardened cement paste. Solutions with dissolved calcium sulfate can react with hydrated cement compounds such as calcium aluminate to form the expansive phase ettringite, which has a much larger molar volume compared to gypsum.

Sulfate attack is controlled by diffusion of sulfate ions into the cement material and is, therefore, lower for cements with lower permeability. Sulfate attack also decreases with the lower tricalcium aluminate content of dry cement powder or clinker. Cements mixed with significant quantities of supplementary cement materials, such as fly ash, ground blast furnace slag, and silica fume, are also less susceptible to sulfate attack. This is attributable to (1) less portlandite available to react with sulfate ions to form expansive gypsum, and (2) reduced permeability of the hydrated cement from pozzolanic reactions with the supplementary cement materials.

*Carbonation*

Carbonation refers to the general process of reaction of gaseous carbon dioxide (or dissolved carbonate originating primarily from atmospheric carbon dioxide) with cement, resulting in the precipitation of calcium carbonate in the form of calcite. Carbonation can lead to physical and chemical consequences, including carbonation shrinkage; however, carbonation shrinkage characterized by surface micro-cracking is unlikely to be continuous or lead to a disruption of the cement matrix. Carbonation can also have a beneficial impact, because calcite has a greater molar volume than portlandite and can lower the permeability of the cement matrix. Detrimental impacts of carbonation are typically associated with corrosion of reinforcing steel (discussed next) due to the reduction in pH. Following carbonation, the pH of the system is controlled by the solubility of calcite with a pH of around 8.3, compared to a pH of around 12.5 before carbonation. At the lower pH following carbonation, the passive oxide layer on the steel surface is unstable, thereby allowing corrosion to occur if oxygen and moisture are present.

*Corrosion of Reinforcing Steel*

Corrosion of reinforcing steel embedded in cementitious materials used for isolating residual radioactivity can lead to the formation of expansive phases, degradation, and cracking. In the presence of highly alkaline solutions typical of cement environments, reinforcing steel is typically protected by a passive oxide film. However, breakdown of this protective oxide film can occur through (1) carbonation of the cement leading to a lower pH of the cement pore water in contact with the steel, and (2) localized breakdown of the film by accumulation of chloride ions at the surface of the steel. Formation of expansive corrosion products can lead to pressure buildup that exceeds the tensile strength of the cement and can cause cracking, spalling, and delaminating of the material. Although chloride ions are common in nature, and small amounts are intentionally added in the mix ingredients of concrete to accelerate set times, the principal sources of chloride ions that cause problems in concrete are from deicing salts, sea water, and chloride ions in surface runoff.

An initiation time and propagation stage are typically considered when estimating times to corrosion of reinforcing steel. Factors that influence chloride-induced corrosion include (1) the type of cement, (2) water-to-cement ratio, (3) sulfate content, (4) service or environmental conditions, (5) degree of carbonation, and (6) properties of the steel. Although critical chloride threshold values have been proposed, due to the multitude of factors influencing the critical threshold, a more broadly accepted methodology is modeling of diffusion of carbon dioxide into the cement materials to estimate the initiation time for carbonation-induced corrosion.

Once the passive film on the reinforcing steel breaks down, the corrosion process can progress. Corrosion rates are a function of (1) permeability, (2) moisture content, (3) temperature, and (4) availability of oxygen. Climate also influences the time to degradation of the material caused by steel corrosion, with more humid environments being more corrosive. In the case of chloride-induced corrosion, oxygen diffusion models have been used to estimate corrosion rates. Rates for carbonation-induced corrosion are typically on the high side, approaching a few mils per year (1 mil= 0.0254 mm=0.001 inches (in.)).

*Alkali-Aggregate Reactions*

Alkali-aggregate reactions can lead to the formation of gels that imbibe water and swell, resulting in expansion, cracking, and degradation of the cement. Alkali-aggregate reactions are usually internally contained in concrete and are not dependent on the diffusion of an aggressive solution into the cement-based material. A source of water is required for appreciable amounts of swelling to occur. Almost all aggregates react to some extent with alkalis in cement. It is

when the reaction results in the formation of expansive products (e.g., gypsum) that serious cracking of the concrete occurs. Expansive alkali-aggregate reactions are known to occur when siliceous (i.e., alkali-silica reactions or ASR) and dolomitic (i.e., alkali-carbonate reactions) limestone aggregates are used. In addition, the rate of expansive reaction is influenced by the size of the aggregates. ASR is the most common, with the majority of the reported instances in the Western States. Alkali-carbonate reactions have occurred in some Midwestern and Eastern States. ASR has also been observed at the Seabrook nuclear power plant in Seabrook, NH. Granite aggregates used in the concrete mix at that facility led to cracking. ASR is a slow process, and its occurrence at Seabrook became evident only decades after the plant was constructed. Short-term reactivity tests (ASTM C1260-14, "Standard Test Method for Potential Alkali Reactivity of Aggregates (Mortar-Bar Method)," issued August 2014) are unlikely to predict the occurrence of ASR over long periods of performance of hundreds to thousands of years.

ASR can be mitigated by selecting aggregates that are not susceptible to deleterious reactions with alkalis in cement. These aggregates are well known and listed in standards and guidance. Because water is also necessary for the reaction to occur, drying the cement materials and preventing future contact with water is also an effective method of mitigating reactions.

*Leaching*

Buried concrete, in contact with percolating subsurface water, can undergo deterioration by the dissolution of the common constituents of cement paste (e.g., portlandite and C-S-H) leading to increased porosity and permeability and the subsequent loss of strength of cement materials. The rate and extent of the leaching are dependent on the acidity of the water, with groundwaters below a pH of 5.5 potentially leading to severe attack. Characteristics of particularly aggressive groundwater with respect to cement leaching have been defined empirically using correlations with dissolved carbon dioxide ($CO_2$), bicarbonate ion, and calcium ion concentrations. Studies have shown that a wide range of acidic, $CO_2$-containing waters have the ability to significantly enhance dissolution of cement phases, compared to pure water. The potential leaching capabilities of subsurface water has also been related to the Langelier Index. The Langelier Index is related to the total dissolved solids, total alkalinity, pH, and calcium content of the water. A positive index indicates that calcium carbonate will be precipitated, while a negative index indicates lime-deficient water capable of dissolving calcium from the hardened cement paste.

The extent of leaching depends on the quality of the cement, water infiltration rates, temperature, and the chemistry of the groundwater. Higher quality cement and decreased water flow through the system will help lessen the detrimental impacts of leaching on cementitious materials.

*Freeze-Thaw Attack*

Damage to cement-based materials occurs when water-saturated concrete is exposed to prolonged cycles of freezing and thawing. Structures most susceptible to freeze-thaw damage are surfaces of structures where flowing or ponding water can remain in contact with the concrete structure for extended periods of time. Several precautions should be taken to avoid freeze-thaw damage, including the following: precluding ponded water from the concrete structure, incorporating entrained air, placing properly, consolidating, and curing. Cementitious materials located at depth are also less susceptible to freeze-thaw damage.

*Microbiological Attack*

Sulfate-producing bacteria are capable of oxidizing elemental sulfur and sulfides to sulfuric acid under aerobic conditions, which in turn degrade cementitious materials. Some bacteria can attack cement-based materials by transforming ammonia into nitrites or nitrates or by producing lactic acid or butyric acid. In the normal design life (e.g., 40 years) of conventional cement-based materials, bacterial action does not seem to be a major cause of deterioration. However, when the chemical durability of cementitious materials must be relied on for hundreds of years, the impacts of bacterial activity should be assessed, although they may be difficult to predict.

*Cracking of Concrete*

Cracking can originate within concrete due to a number of mechanisms. During placement, if the evaporation rate is great enough, the concrete surface can develop tensile stresses sufficient to crack the concrete. These plastic-shrinkage cracks typically extend through the entire concrete member. Cracking can also be caused by (i) settlement of the concrete member, (ii) flexural stresses, and (iii) thermal effects. The continued removal of water by the hydration process will generate a chemical-shrinkage stress that can initiate autogenously shrinkage cracks. Subsequent drying due to ambient conditions will also create shrinkage stress that generates drying-shrinkage cracks. Absent environmental conditions that may cause concrete-material degradation, cracking can be the most severe degradation mechanism affecting concrete. Transport through cracks in concrete will only be of consequence if the cracks extend throughout the concrete member. Relatively large amounts of water can be transported through a crack, depending on the total potential water-head across the crack, as well as the crack aperture and density.

*Degradation of Engineered Covers*

Engineered surface covers are commonly designed to eliminate or significantly limit infiltration of subsurface water into the waste but may also be designed to limit gaseous release, provide shielding, reduce the likelihood of contact with the waste, or create physical stabilization at a site. From an infiltration management perspective, engineered covers are based on either resistive or water balance concepts. Covers based on resistive concepts use impermeable layers to prevent water from contacting the waste. Evapotranspiration covers based on water balance concepts attempt to mimic natural systems to evaporate and/or transpire water from the system, typically by allowing plants to grow on the cover.

*Degradation of Resistive Covers*

Resistive engineered cover designs have a wide range of configurations, ranging from a simple one-layer cover to complex multilayered designs composed of soils and geosynthetics. Soil materials can include vegetative soils, permeable sand and gravel drainage layers, low hydraulic conductivity clay soils, and filter soils to preclude the migration of fines from soils

overlying drainage layers, which cause the drainage layers to clog. Geosynthetic materials include geomembranes and geosynthetic clay layers. Composite barriers use both soils and geomembranes/geosynthetic clay layers. Geomembranes are essentially impermeable polyvinyl chloride (PVC) or polystyrene layers, while geosynthetic clay layers are composed of a thin layer of bentonite between two geosynthetic textiles, which may be used in conjunction with geomembranes. The effectiveness of these engineered covers lies in the ability to significantly limit infiltration of subsurface water into the waste. Cover performance is affected by waste stability, the degree of settlement, and slope stability of the cover system.

Degradation mechanisms unique to engineered cover materials, other than those specific to inadequate design or construction and QA/QC issues, are discussed in NUREG/CP-0195, "Proceedings of the Workshop on Engineered Barrier Performance Related to Low-Level Radioactive Waste, Decommissioning, and Uranium Mill Tailings Facilities," held August 3–5, 2010. Degradation mechanisms include the following physical, chemical, and biological processes:

- physical processes such as freeze-thaw, wet-dry cycle, differential settlement, retention of borrow soil structure (peds or clods) during construction, silting-in of drainage layers, ultraviolet degradation, thermal degradation, erosion, fire, and pedogenesis

- chemical processes, such as oxidation of geosynthetic materials and cation exchange mechanisms in sodium bentonites

- biological processes that involve unanticipated ecological consequences of designs that, by creating habitat for deep-rooted plants, burrowing animals, and soil microorganisms, can alter soil hydraulic properties

NUREG/CR-7028 assessed changes in hydraulic conductivity of 27 cover soils from a variety of sites across the United States. A major finding of the report was that as-built properties of cover soils changed significantly over a period of 5 to 10 years regardless of climate or design. For the covers that were evaluated, measured saturated hydraulic conductivities increased regardless of the initial saturated hydraulic conductivity. Saturated hydraulic conductivity of storage and barrier layers for in-service covers was always at least ten times greater than as-built values. In a few cases in-service values were thousands of times greater. All evaluated cover types showed alterations in saturated hydraulic conductivity in all climates and for both barrier and storage layers. Changes in hydraulic conductivity were primarily attributed to cracks formed by drying events.

*Compacted Clay Barrier*

The function of the clay barrier layer in the engineered cover is to prevent and block infiltration of subsurface water through the cover into the waste. The compacted clay layer is frequently specified to have a saturated hydraulic conductivity of $1x10^{-7}$ cm/sec or less. The longevity and effectiveness of the engineered cover are influenced by the ability of the clay layer to retain low permeability characteristics. However, laboratory and field studies have shown that compacted clay can develop, even within ten years, distinct soil structures such as aggregates and planes of weakness due to pedogenic processes such as wet-dry and freeze-thaw cycling. These soil structures, in turn, can significantly increase the hydraulic conductivity of the initial compacted clay. Larger changes were observed for soils with lower as-built saturated hydraulic conductivity and soils with a greater proportion of clay particles in the fines fraction. Degradation of barriers is related to grain size of the material with higher clay content materials undergoing greater increases in hydraulic conductivity (NUREG/CR-7028). Unanticipated

ecological processes (e.g., biointrusion), compaction, a drier or wetter than optimum environment, freeze-thaw cracking, and differential settlement can result in the degradation of clay and other resistive barriers.

*Drainage and Filter Soil Layers*

A common degradation mechanism affecting these drainage and filter components of the engineered cover system is the potential clogging of these materials by finer soil particles from overlying soils, including colloidal or biological materials. Clogging of drainage and filter soil layers can greatly reduce their permeability, allowing more water to move downward through waste material, and render them unable to perform their intended function to drain subsurface water.

*Composite Soil Covers*

Composite covers comprise a combination of a compacted clay layer, geosynthetic clay layer, or geomembrane. In general, composite covers have performed well in the time frame of available field studies (10+ years). Geomembranes protect clay barriers and geosynthetic clay layers by eliminating or significantly reducing vertical water vapor migration. Geomembranes (e.g., HDPE) are often placed on top of geosynthetic clay layers to extend the performance life of the geosynthetic clay layer. Degradation mechanisms of geomembranes include: puncture by granular soils and construction equipment; behavior of "waves" or fabric wrinkles due to temperature and overburden stresses; long-term degradation of geomembranes under the influence of ultraviolet light, chemicals, and radiation effects; the potential for slippage between geomembranes and adjacent materials; material embrittlement over time; and animal burrowing and root penetration. Some degradation mechanisms, such as exposure to ultraviolet light, can be managed with effective QA/QC.

Geosynthetic clay layers are sometimes used instead of compacted clay and seem to perform better with respect to water flow, chemical degradation due to cation exchange, and mass transport (i.e., diffusion and retardation). As with geomembranes, however, geosynthetic clay layers have inherent problems due to installation activities (e.g., puncturing and degradation by construction equipment). Geosynthetic clay layers can experience desiccation cracking similar to clay layers. In addition, recent research seems to indicate a potential for cation exchange between commonly available calcium-laden fluids and the sodium in the geosynthetic clay layer bentonite, thus rendering the geosynthetic clay layer incapable of functioning as a low-permeability barrier layer in engineered-cover systems (James et al., "Field Performance of GCL under Ion Exchange Conditions," issued 1997; Melchoir, "In-Situ Studies on the Performance of Landfill Caps," issued 1997; Lin and Benson, "Effect of Wet-Dry Cycling on Swelling and Hydraulic Conductivity of GCLs," issued 2000). Research is being conducted to more carefully study these degradation mechanisms.

NUREG/CR-7028 evaluates the engineered properties of a variety of covers located around the Unites States, including those that used composite materials. Under certain conditions, composite covers were found to perform better than conventional soil covers. For example, geosynthetic clay layers hydrated to water contents in excess of 50 percent retained relatively low hydraulic conductivity over the 5–10-year timeframe of the study. To achieve these water contents, NUREG/CR-7028 recommends geosynthetic clay liners (GCLs) be placed on subgrades with moisture content greater than 10 percent with an overlying geomembrane to prevent desiccation.

*Degradation of Evapotranspiration Covers*

Evapotranspiration covers attempt to manipulate the water balance of the source zone by enhancing soil water storage and evapotranspiration by planted or wild vegetation. The performance of evapotranspiration covers depends on many factors, especially the climatology, soil hydrology, and plant ecology at a site. Evapotranspiration covers may be used in a variety of settings but may be most effective in arid or semiarid climates with high potential evapotranspiration.

The effectiveness of an evapotranspiration cover is dependent on the development of a design that is effective over the range of expected natural and ecological conditions, which are inherently variable over the timeframe of most decommissioning analyses. As discussed in Section P.1.4, evapotranspiration covers have been analyzed in detail at a number of sites. With effective design and development, evapotranspiration covers may be very effective, especially in arid and semiarid climates. In humid climates, evapotranspiration covers may be effective at managing a substantial fraction of the infiltration but not achieve design goals. Infiltration can commonly exceed evapotranspiration in humid climates or in colder climates, where a large fraction of infiltration may occur as snowmelt when evapotranspiration is low. Therefore, one of the major lessons learned, albeit not related to physical degradation, is that the design of an evapotranspiration cover must consider natural and ecological variability over the period of performance.

Degradation mechanisms associated with evapotranspiration covers include unanticipated ecological consequences, evolution of soil properties (e.g., pedogenesis), and effects of disturbance (e.g., fire, land use) on plant ecology. An example of unanticipated ecological consequences is the development of deep-rooted plant species that result in pathways for moisture below the design zone for moisture storage and removal.

*Degradation of Erosion Protection Barriers*

The degradation of erosion protection barriers is described in detail in the example provided in Section P.2.

*Degradation of Permeable Reactive Barriers*

Permeable reactive barriers are *in situ* constructed walls below the land surface that intercept contaminated groundwater, which is funneled through it. Reactive materials in the wall can sorb chemicals and radioactive species on their surfaces and precipitate contaminants dissolved in the flowing water. In some cases, nutrients and oxygen in a permeable reactive barrier help microbes in the soil to precipitate contaminants and radioactive species. Experience with permeable reactive barriers seems to indicate that not all of them are performing well, often due to poor placement in the groundwater flow field. Material properties, such as grain size of reactive zeolites, can be changed in the construction process, thereby reducing the hydraulic conductivity. There seems to be a decrease of barrier performance due to loss of reactivity and permeability over relatively short periods of time. Systems have experienced challenges associated with diversion of water around the permeable reactive barrier caused by decreased permeability of the reactive barrier material in less than 5 years.

*Degradation of Vertical Barriers*

Various types of subsurface vertical barriers are in use. Their primary purpose is to impede or preclude horizontal groundwater flow. These vertical barriers are placed at depths up to 60 m (200 ft) and often vary in thickness from 0.6–1.2 m (2–4 ft). The barriers must extend down to an impermeable natural horizontal barrier, such as a clay zone, to effectively impede groundwater flow from below. These barriers are often designed as temporary or semi-permanent remediation techniques to isolate contaminated fluids from migrating to

uncontaminated surrounding groundwater. Some soil-bentonite mixtures are not able to withstand attack by chemicals such as strong acids, bases, salt solutions, and certain organic chemicals. This hastens the deterioration of the barrier. Verification that the vertical wall forms a continuous barrier is critical to the function of this technology. Although it may be difficult to identify flaws in the continuity, or gaps in the wall, monitoring is essential to verify their performance. Although vertical walls have been used for decades, the process of designing the proper mix of wall materials to contain specific contaminants is less well developed.

*Potential Levels of Functionality and Uncertainty*

This section gives some general information to help licensees initially consider the use of engineered barriers. As discussed in Sections P.1.2 and P.1.3, licensees are responsible for developing acceptable technical bases and conducting analyses of engineered barriers proposed for a specific site.

The ranges of functionality or performance for different barriers and the associated uncertainty are based on a broad consideration of observations and analyses throughout the national and international community. Potential ranges of functionality are levels of performance that can likely be supported by technical bases and analyses, not solely based on demonstrated field experience. As the time scales get longer, past direct observation of the performance of engineered barriers that can be cited as a basis becomes less likely, and therefore, performance for longer times becomes more uncertain and based more on inference. The functionality provided here can be thought of as the level of performance believed to be reasonably achievable with proper design, analysis (Section P.1.2), technical basis (Section P.1.3), and implementation (quality), given current understanding and engineering practice. The ranges for potential functionality help provide direction as to when a less technical basis may be needed (assume less than typical performance) compared to a more technical basis (credit is taken for more than typical performance). The level of uncertainty should be considered in developing a monitoring and maintenance plan. Barriers with less uncertainty might need less reliance on monitoring and maintenance. In contrast, barriers with higher uncertainty may need substantial monitoring and maintenance until uncertainties are reduced.

The NRC's discussion in this section is an initial attempt to provide some insights on potential functionality and uncertainty, using readily available information. This section could be expanded, based on future studies and input from other programs involved with engineered barriers. Typical levels of functionality or performance of the main engineered barriers are as follows.

*Cement-Based Engineered Barriers*

The performance of cement-based materials to isolate residual radioactivity can be divided into two categories:

- hydrologic effectiveness or physical containment of the wastes to preclude water contacting the waste

- chemical effectiveness or the ability of the high pH characteristics of the intact and degraded concrete to limit transport of the radionuclides to the accessible environment

Absent environmental degradation factors of concrete (e.g., sulfate attack, chloride corrosion), full-depth cracking of the concrete member can be the most severe degradation mechanism

causing contact of water with the waste and the subsequent release of radionuclides. Accordingly, cement-based physical barrier structures need to be monitored for hydrologic effectiveness, and the projected service life for the structure should be revised based on an analysis of the monitoring data. Assuming adequate design and construction practices and excellent QA/QC, followed by a competent monitoring program, a service life of tens of years to a few hundred years appears feasible.

In addition to a physical barrier, Portland cement-based materials provide a chemical barrier by raising pH of water moving through it. For some radionuclides this can substantially lower their solubility and reduce releases. Some cementitious materials (e.g. HLW tank backfill grouts) are designed to provide chemically reducing conditions that can also limit solubility and transport of certain elements. The length of time chemical barriers, which mitigate releases of radioactivity, may need to perform to demonstrate compliance with radiological criteria for license termination is dependent on the length of time residual radioactivity poses a risk to potential receptors. Chemical barriers to waste release may perform for longer time periods than hydrological controls. Nonetheless, the longevity of chemical effectiveness is strongly related to the bulk hydraulic properties of the material, the chemistry of infiltrating water, and the composition of the cementitious materials (e.g., reductive capacity and cement fraction). This is because the quantity of water flowing through the cement matrix, the chemical gradients between infiltrating water and cement pore water, and the reductive and buffering capacity of the cementitious materials dictate the rate at which cement components that aid in the retention of radioactivity are leached from the system.

A cementitious barrier used to limit potential intruder contact with waste, with proper design, construction, and QA/QC, could be expected to be effective for hundreds of years if it remains unexposed to aggressive environmental conditions (e.g., high sulfate, excessive freeze-thaw cycles). Performance of this type of barrier may be enhanced with appropriate monitoring, repair, and remediation strategies.

*Engineered Covers*

Research indicates that extensive desiccation of clay barriers in soil covers has compromised the ability of clay barriers to retain low permeability characteristics and prevent infiltration of water through the cover and into the waste (Albright, “Field Water Balance of Landfill Final Covers,” issued 2004). Conversely, composite covers composed of a combination of compacted clay buried at sufficient depth, geomembranes, and GCLs have performed well in the timeframes of available field studies (10+ years). Continued monitoring may be needed to verify the effective lifetimes of these covers. Current experience provides evidence that hydrologic functionality of tens of years for composite covers appears to be feasible. Longer hydrologic functionality may be feasible with the proper development and implementation of the elements provided in the previous sections, and with continued research. Existing uncertainty in long-term functionality could be reduced by additional technical bases, analyses, testing, and field experience. Geomembranes do not have the experience base of common natural materials or a man-made material like cement, which has been used for hundreds to thousands of years. Therefore, until the experience base is developed, a cautious approach is needed for the long-term performance of novel materials in engineered barriers.

Strategies listed in NUREG/CP-0195 for minimizing the negative impacts of degradation processes include (1) paying careful attention to construction QA because it is especially important to the successful short-term performance of the cover, (2) identifying the processes that have the greatest impact on performance, (3) understanding the total system and planning accordingly, (4) analyzing each component within context, and (5) identifying and standardizing

exposure scenarios that ultimately control the performance objectives and cover features related to these objectives.

Unintended consequences must be considered when trying to diminish one form of degradation. The licensee must clearly determine the function of each cover component to diminish the risk of unintended consequences. For example, the use of plants to remove moisture from the barrier and inhibit movement of water down through the waste also results in enhanced radon transport upward through the barrier (NUREG/CP-0312 (Fuhrmann et al., 2018)). The intended function of each component must be clearly defined, in addition to the potential risks to the primary function. This will assist in delineating pathways for managing the total system performance.

Designers and analysts should recognize that soil properties may change quickly and therefore should minimize the consequences of these changes by designing and constructing covers for sites with long-lived radionuclides that mimic longer-term conditions that are congruent with nature. The resources required to maintain the engineered system must also be considered. Instruments can help identify the timing of a degradation process. There is a need to develop techniques that seek to understand magnitude and direction of natural changes anticipated to occur. One approach is to develop a catalog of natural analogs.

Evapotranspiration covers have been studied in the field on large scales, as discussed in Section P.3. These covers have performed well in arid and semiarid climates when they are well designed, constructed, and monitored. Because they attempt to mimic natural processes, the level of performance achieved over the long term would be expected to exceed that of resistive infiltration barriers. Whereas a conventional (resistive) cover is subject to failure by fast pathways (likely to be more discrete; either it is functioning, or it is not), an evapotranspiration cover is more likely to have different degrees of performance (likely to be more gradual, based on exceeding the design capacity for water management). Comparisons of evapotranspiration covers and resistive covers discussed in NUREG/CP-0312 show that evapotranspiration covers perform well in the field on the order of 10 years, and in theory, may have some passive performance over the very long term.

*Erosion Protection Barriers*

The effectiveness of engineered cover components developed for erosion control could exceed 1,000 years. However, erosion control designs may not be adequate to preclude excessive infiltration. For example, deposition of windblown fines into openings between rocks on rip-rap covered sites may result in plants taking root and disrupting the clay barriers under the rock. Appendix Q contains a detailed example, including some of the design considerations.

Aside from the depth to waste, most engineered covers would not provide a substantial barrier to common practices assumed in intruder analysis (e.g., home construction[2], well installation). In some cases, the licensee could present arguments on the likelihood that a well driller would intrude into an engineered system comprised of large rock for erosion control or systems that rely on significant quantities of cementitious materials to isolate the radioactivity. For instance, drillers accustomed to drilling into unconsolidated materials may be alerted to a hazard if cementitious materials are discovered near the surface.

*Permeable Reactive Barriers*

[2] Relatively thick covers may mitigate the impact of basement construction, with basements typically assumed to be dug to a depth from ground surface of 3 m, if erosion does not significantly decrease the cover thickness.

There has been limited (approximately 15 years) experience with permeable reactive barriers. Effective lifetimes of these barriers, from the literature, appear to be limited to less than 10 years.

*Vertical Barriers*

As noted previously, it is difficult to identify flaws in the continuity and gaps in constructed vertical walls. In addition, some of these walls are not able to withstand chemicals such as strong acids, bases, and certain organic materials. Moreover, the process of designing a proper mix of wall materials to contain specific contaminants (hazardous chemicals and radionuclides) is less well developed. Accordingly, the effective service life of these structures would likely range in the low single digits of years and their performance should be demonstrated by field testing.

## P.2 Example of a Graded Approach for Erosion Protection Covers

The following text provides a detailed example of the application of a graded approach to the design and implementation of engineered barriers for erosion protection. The graded approach can be readily used for the design of an engineered barrier, specifically a soil or rock cover that is provided for erosion control. This approach provides significant flexibility and can be adapted to design covers at a wide variety of sites with a wide range of waste inventories.

The criteria provided in NUREG-1623, "Design of Erosion Protection Covers for Long-Term Stabilization," issued September 2002, can be used for lower and higher risk decommissioning sites. NUREG-1623 has been used by NRC staff to review and approve erosion protection designs at over 40 different uranium mill tailings impoundments.

Erosion control covers for uranium mill tailings impoundments provide examples of design and construction of robust engineered barriers for long-term protection of radioactive materials. Requirements for engineered barriers at uranium mill tailings sites are set by the Uranium Mill Tailings Radiation Control Act (UMTRCA) of 1978, 42 U.S.C. § 7901 et seq.; the U.S. Environmental Protection Agency's implementing regulations at 40 CFR Part 192, "Health and Environmental Protection Standards for Uranium and Thorium Mill Tailings;" and the NRC's implementing regulations in 10 CFR Part 40, "Domestic Licensing of Source Material," Appendix A, "Criteria Relating to the Operation of Uranium Mills and the Disposition of Tailings or Wastes Produced by the Extraction or Concentration of Source Material from Ores Processed Primarily for Their Source Material Content." Under these regulations, erosion control covers are required to be designed to remain effective for up to 1,000 years without reliance on ongoing active maintenance. Although UMTRCA also requires State or DOE ownership and long-term care of the uranium mill tailings sites, including maintenance of the covers as needed, the covers are designed to function independently of maintenance. Therefore, the monitoring or surveillance and maintenance provided by DOE can be thought of as a backup to the robust design, or a defense-in-depth approach to long-term protection. Over 25 years of experience is available, including the NRC's guidance and technical basis for the design of robust erosion control covers and DOE's construction of these covers and monitoring of their performance. This program offers an approach and lessons learned for one kind of robust engineered barrier that could have some application to the design of other types of robust barriers. Although the graded approach to engineered barriers in this decommissioning guidance offers greater flexibility to select appropriate options for engineered barriers (including reliance on maintenance), the UMTRCA covers offer an excellent example of a robust erosion control barrier to maintain stability for sites with a long-term hazard.

Robust engineered barriers for erosion control should be designed, but a graded approach to the design should be taken with respect to selection of the design flood, evaluation of rock durability, and selection of appropriate design factors. Each of these three areas is described below.

### P.2.1 Selection of Design Flood

One of the phenomena most likely to affect long-term stability is surface water erosion. To mitigate the potential effects of surface water erosion, the staff considers it very important to select an appropriate rainfall event on which to base the erosion protection designs. Further, the staff considers that the selection of a design flood event should not be based on the extrapolation of limited historical flood data, due to the unknown level of accuracy associated with such an extrapolation. The probable maximum precipitation (PMP) is computed by deterministic methods (rather than statistical methods) and is based on site-specific hydrometeorological characteristics. The PMP has been defined as the most severe reasonably possible rainfall event that could occur as a result of a combination of the most severe meteorological conditions occurring over a watershed. No recurrence interval is normally assigned to the PMP; however, the staff has concluded that the probability of such an event being equaled or exceeded during a 1,000-year stability period is very low. Accordingly, the NRC staff considers the PMP to provide an acceptable design basis.

The probable maximum flood (PMF) is based on the occurrence of the PMP and is considered to represent the most severe flood that can reasonably be expected to occur over a particular drainage basin.[3] There is no probability assigned to the PMF, but the staff will generally not accept the use of statistically derived floods when the analysis time period significantly exceeds recorded data, due to the unreliable extrapolation of flood records based on short-term data. NUREG-1623 contains further discussion of the use of the PMP and PMF.

### P.2.2 Rock Durability

Rock durability is defined as the ability of material to withstand the forces of weathering. Primary factors that affect rock durability are (1) chemical reactions with water, (2) saturation time, (3) temperature of the water, (4) scour by sediments, (5) windblown scour, (6) wetting and drying, and (7) freezing and thawing.

For rock to remain effective in controlling erosion, the rock size selected and emplaced should not be reduced in size by weathering processes. Therefore, if the rock size used for the cover does not diminish over the 1,000-year compliance time period, its ability to control future erosion will be sustained. However, uncertainties exist with estimating future rock durability. For example, quantitative studies of weathering rates of different rock types and minerals are limited, in general, and not expected to be available for specific rock sources that might be selected by licensees. As a result, three evaluations of rock durability should be conducted to provide multiple and complementary lines of evidence and greater confidence in the future durability of the rock source selected. These evaluations are (1) rock durability testing and scoring, (2) absence of adverse minerals and heterogeneities, and (3) evidence of resistance to weathering. A description of each of these evaluations follows.

[3] Although consideration of the PMP is expected to provide a bounding analysis for the PMF in most cases, if other features, events or processes (FEPs) could lead to greater impacts, then those FEPs should also be considered.

### P.2.3 Rock Durability Testing and Scoring

NUREG-1623 includes a procedure for determining the acceptability of a rock source, which the NRC staff developed and used for selecting durable rock for erosion covers for uranium mill tailing sites. This procedure provides a consistent and quantitative way to evaluate rock sources at NRC-licensed sites using standard parameters that are good indicators of rock durability. Test samples of the selected rock source should be representative of the specific rock expected to be used. The number and location of the samples should be determined based on the variability of the rock source, such as, texture and mineralogy that could affect the individual test results. The test procedure generally includes the following:

- Test results of four parameters (specific gravity, absorption, sodium sulfate, and Los Angeles abrasion) from representative samples are scored on a scale of 0–10.
- The score is multiplied by a weighting factor, which focuses the scoring on those tests that are the most applicable for the particular rock type being tested.
- The weighted scores are totaled, divided by the maximum possible score, and multiplied by 100 to determine the rating.
- The rock quality scores are then compared to the criteria that are measures of acceptability.

After these tests are conducted, an overall rock quality score is determined. Rock scores of 80 percent or greater indicate high-quality rock that can be used for most applications. Rock scores between 65 percent and 80 percent indicate less durable rock that can also be used for most applications. Rock scoring less than 65 percent cannot be used for critical areas, such as diversion ditches or poorly drained toes and aprons. Rock scoring between 50 percent and 65 percent can be used in noncritical areas, such as well-drained side slopes, provided the rock is oversized. Rock scoring less than 50 percent is not recommended for use in any application. NUREG-1623 contains additional discussion of specific tests and the scoring procedures. For example, Table F-1 in NUREG-1623 lists a number of American Society for Testing and Materials (ASTM) standards which should be used for quality and gradation tests.

### P.2.4 Absence of Adverse Minerals and Heterogeneities

Licensees should provide results of petrographic analyses of the selected rock source and available published data that support the absence of adverse minerals that could cause rapid degradation of the rock, such as clays, olivine, or calcite cement. If adverse minerals are present, they should evaluate the potential effect on future weathering of the selected rock.

Licensees should place particular emphasis on selecting rocks that do not have appreciable clay content or do not contain minerals that could rapidly weather to clays. The staff examined the causes of rock durability failures in typical applications, such as placement on dam slopes or stream channels. In most cases, the staff determined that durability failures were caused by the presence of expanding clay-lattice minerals, which, when exposed to moisture or freeze and thaw cycles, caused the rapid deterioration of the rock.

Licensees should identify either the absence or presence of heterogeneities that could adversely affect the selected rock source, such as clay or shale partings, interbeds, fractures, alteration zones, or vein deposits. Such heterogeneities can be zones of water flow and

associated chemical alteration or zones of breakage due to freeze and thaw. Licensees should characterize heterogeneities present and discuss the ability to avoid these adverse heterogeneities when removing the selected rock for use. They should also evaluate how heterogeneities could affect achieving the acceptable size of the selected rock. Breakage along thin interbeds or fractures during rock quarrying, transport, or emplacement could result in reduction of the rock size necessary for erosion protection.

### P.2.5 Evidence of Resistance to Weathering

Licensees should provide direct evidence from the selected rock source, such as minerals that are resistant to weathering, resistant cements, and regional or local geomorphic evidence of slow weathering of the selected rock that outcrops in other locations. Examples include rounded boulders or thin weathering rinds. Weathering rind thickness and alteration of minerals and rock properties from exposures of the weathered selected rock source could provide insights on the extent and nature of future weathering of the selected rock source from fresh quarry exposures. Identify and describe any available studies of weathering rates of the selected rock source.

Licensees can also use indirect evidence from other locations where the general rock type is similar to the selected rock source. For example, evidence of durability from a diabase igneous rock found in Europe could be used to provide insights on a diabase rock source in Maryland, because the general mineralogy of diabase is similar, regardless of the location. This approach allows the use of datable natural or archaeological/historical rock sites (called analog sites) that could provide general evidence on rock weathering rates or time periods during which rock types have remained resistant to weathering. For example, as mentioned in Section P.3.3, numerous datable archaeological sites, such as Stonehenge (constructed about 4,000 years ago of diabase and silica cemented sandstone), Hadrian's Wall (constructed by the Romans over 2,000 years ago of primarily diabase), and numerous buildings, monuments, and megaliths in Europe, could be used to demonstrate that these rock types have been resistant to weathering over time periods that exceed the 1,000-year period of regulatory compliance for a robust erosion cover. Historical evidence can also provide useful insights on the durability of certain rock types. One example is the comparison of dated Civil War photographs of diabase outcrops in Devil's Den at the Gettysburg Battle Field Park to present-day conditions of the same outcrop. Such a comparison indicates that this diabase has been resistant to weathering for about 150 years. Similarly, dated grave markers or historical buildings made from the selected rock source or similar rock type can provide evidence of resistance to weathering for 100–200 years. Appendix A of NUREG/CR-2642, “Appendix to Long-Term Survivability of Riprap for Armoring Uranium Mill Tailings and Covers: A Literature Review,” 1982, provides additional information on rock weathering, durability, and examples of analogs that contain insights on general weathering rates of various rock types.

As previously noted, natural rock sites, such as age-dated glacial erratic, could also be used to show that the particular rock type has been resistant to weathering for over 10,000 years. Examples that illustrate this approach include age-dated diabase erratics from New York and North Dakota that are over 10,000 years old, and river scour features in diabase that have been preserved for about 20,000 years in Pennsylvania.

For each of the above evaluations, the licensee should provide the appropriate supporting data, information, and references and identify the location of the selected rock source, along with regional and local rock source descriptions.

### P.2.6 Selection of Appropriate Design Factors

In selecting appropriate input parameters for calculating erosion protection size and thickness, it is important to choose values that reflect the degree of risk and the importance of the rock layer, as it contributes to overall stability. However, the selection of many input parameters to various models can sometimes be subjective and will need to be based largely on engineering judgment. Where there are large ranges in values, or where a parameter cannot be well defined, or is not well known, it has been the general policy of the NRC staff to accept the use of reasonable ranges and distributions of input parameters. For well-known or accepted parameters with narrow empirical distributions or very narrow ranges, expected values should be used as appropriate. For less well-known parameters, such as those based on little empirical data or with broad distributions, conservative values should be chosen from within the observed distributions or estimated range. In any case, there should a reasonable and defensible technical basis for the choice of a design-basis event or design criteria, and the staff will accept values that can be justified. Otherwise, reasonably conservative values will be needed.

### P.2.7 Erosion Cover Analysis Process

At a lower risk site, where engineered barriers are needed to meet applicable requirements for only about 100 years and there is a lower hazard level should institutional controls and maintenance fail (up to the public dose limit of 1.0 mSv/y (100 mrem/y)), the principal design basis and goal is to ensure the relative stability of the contaminated material by providing a cover design that maintains control of the material. Control is achieved by providing a relatively robust design that prevents offsite movement (e.g., erosion by natural forces) of the material. The rock erosion protection barrier could protect a second layer of material; for example, one that might be a radiation shielding barrier or an infiltration barrier, depending on the radionuclides at the site and natural processes important to achieving compliance with the dose criteria.

The design should be able to survive the occurrence of relatively rare events, and the erosion protection should be sufficiently robust to remain effective for about 100 years. Using the guidance and rationale contained in NUREG-1623 for a 100-year stability period, the barriers should be designed to resist a flood equivalent to either the regional historic flood of record or about half of the PMF, whichever is greater. The licensee should provide rainfall, flood, and erosion analyses that justify the design. A design that meets the suggested flooding and erosion protection criteria of NUREG-1623 is acceptable. The rock itself should be sufficiently durable to remain effective for at least 100 years by obtaining a rock quality score of at least 65. The computations and selection of input parameters should rely on reasonable and justified estimates.

For a higher risk site, with long-lived radionuclides, or where failure of institutional controls and maintenance could result in a higher hazard of 1.0–5.0 mSv/y (100–500 mrem/y), the principal design basis and goal is to ensure the control and stability of the contaminated material by providing a robust design that will remain effective for a period of 1,000 years or more by preventing erosion by natural forces. These covers should be designed to maintain control and stability with no reliance on active maintenance. However, monitoring and maintenance will be conducted as a backup to provide defense in depth.

The staff could approve an engineered barrier design that is effective and maintains control of the material for a period exceeding 1,000 years. Using the guidance and rationale contained in

NUREG-1623, the barriers should be designed to resist severe localized rainfall events and large floods on nearby streams. The design rainfall event should be the PMP, and the design flood should be the PMF. A design that meets the suggested flooding and erosion protection criteria of NUREG-1623 is acceptable. The rock quality score should be at least 85, and the selection of input parameters to various models should account for the unknowns associated with a very long stability period and the high-risk site.

For sites like this, if erosion is a significant issue and there are some uncertainties associated with the magnitude of this erosion, the staff will approve a design that would likely incorporate (1) covers designed to resist erosion for a stability period exceeding 1,000 years, (2) a long-term surveillance program that monitors the magnitude and rate of erosion, and (3) sufficient funding for the surveillance, repair, and replacement of some of the erosion protection. The staff will work closely with the expected long-term custodian to determine the amount of funding needed.

It is important to reiterate that the requirements of 10 CFR Part 40 are very prescriptive and may have precluded the use of many types of erosion protection designs. The design criteria suggested in NUREG-1623 may be used at decommissioning sites using approaches different from those used in uranium mill tailings applications. For example, nearly all tailings sites were designed with disposal cell side slopes of about 1 vertical (V) on 5 horizontal (H). Based on the stability of erosion protection placed on much steeper slopes of stream channels, levees, and dam slopes, there is evidence that may support the use of slopes steeper than 1V on 5H for the side slopes of disposal cells. The criteria in NUREG-1623 were not developed for use on specific slopes and may be adapted for steeper side slopes, as necessary. However, minor changes to construction specifications may need to be considered for steeper slopes, such as emphasizing careful rock placement,.

Table P.2 summarizes the application of the graded approach for the design of erosion protection for the sites discussed above.

**Table P.2 Summary of the Graded Approach for the Design of Erosion Protection Systems**

| Level of Risk | Flood Design Basis | Rock Durability Score | Confidence in Selection of Input Parameters |
|---|---|---|---|
| Lower | ½ PMP / ½ PMF | > 65 | Reasonable |
| Higher | PMP/ PMF | > 85 | Very High |

### P.2.8 Technical Basis for Design and Performance of Erosion Protection Covers

The use of the guidance presented in this document and in NUREG-1623 will result in designs that provide acceptable long-term erosion protection at decommissioning sites. These documents contain design criteria that incorporate a strong technical basis, including (1) use of NRC experience and lessons learned at various sites to develop and implement erosion protection guidance, (2) use of appropriately conservative design bases and computational procedures for erosion protection designs, and (3) an extensive archaeological and natural basis for long-term stability.

### *P.2.8.1 NRC Experience and Lessons Learned*

The NRC staff has over 20 years of experience in the design and review of erosion protection covers. This experience includes activities associated with (1) development of design guidance for uranium mill tailings reclamation, including use of technical studies sponsored by the NRC to address specific design problems, (2) review of reclamation plans and DPs at about 50 sites, including review of specific problem sites where challenging erosion problems were addressed, and (3) review and inspections of construction problems and deficiencies. The staff has applied this experience in its review of uranium mill reclamation plans and DPs on numerous occasions.

### *P.2.8.2 Guidance Development*

UMTRCA regulations established design standards to be met at uranium mill sites for Title I (inactive sites) and Title II (licensed active sites), as required by UMTRCA. Specifically, the design standard for long-term stability was established to be 1,000 years to the extent reasonably achievable, or in any case at least 200 years. When the regulations were developed, there was very little experience associated with providing designs that would remain effective for such long periods of time. To address this problem, the NRC staff worked closely with DOE and various contractors in the 1980s to establish design criteria and guidance for long-term stabilization. The NRC held several joint meetings with DOE and DOE contractors to formulate guidance and design procedures to address the long-term stability of erosion protection covers. These procedures were published in various technical and construction documents that were developed by DOE, DOE contractors, and the NRC staff.

During the development of design guidance, it was recognized that some of the procedures normally used for the design of erosion protection were not necessarily appropriate for long-term stabilization or for conditions that would be encountered at various sites. To address such issues, the NRC sponsored extensive technical studies. These studies were conducted in the 1990s at Colorado State University and included prototype flume studies to determine rock sizing procedures for overland flows; flume studies to determine rock sizing and volume requirements for aprons and toes of slopes; gully studies to determine rates, magnitude, and location of gully development on reclaimed slopes; and rock durability studies to address rock weathering rates and tests needed to ensure adequate rock quality. The results of the studies were published in nationally recognized journals and peer reviewed by experts in the field of erosion protection design. One widely used example includes detailed guidance for the design of riprap for flood flows that would be expected down the side slope of a disposal cell (S.R. Abt and T.L. Johnson, “Riprap Design for Overtopping Flow,” American Society of Civil Engineers, Journal of Hydraulic Engineering, 1991 (Abt and Johnson, 1991)). NUREG-1623 includes results from this and other publications.

### *P.2.8.3 NRC Staff Review of Challenging Sites*

Vegetated soil covers, rock covers, and composite covers (soil and rock) were used extensively in the reclamation and stabilization of all uranium mill tailings sites. In general, most of these sites presented no significant erosion problems, and DOE and several NRC licensees had a great deal of success in designing and constructing these covers. However, several sites presented challenging erosion problems. Table P.3 provides examples of several significant erosion problems that were solved with engineered barriers, using design criteria in NUREG-1623.

**Table P.3 Examples of Engineered Barriers Used to Solve Erosion Problems**

| Site | Erosion Problem | Solution to Problem |
|---|---|---|
| Maybell UMTRCA Site | Downstream gullies that could scour to a depth of about 20 feet could encroach on the disposal cell. Unstable local base levels could cause further increase in the potential for gullying. | DOE provided extensive rock aprons to check erosion advance to the disposal cell, using criteria for scour depth and rock sizing provided in NUREG-1623. |
| Grand Junction UMTRCA Site | Deep gullies with relatively large drainage areas existed in the approximate center of the proposed disposal cell. Hydraulically steep slopes were present near the toe of the cell. | DOE provided diversion channels and channel outlets with very large riprap to safely convey flows around the disposal cell, using criteria found in NUREG-1623. |
| Rifle (Estes Gulch) UMTRCA Site | The disposal cell was excavated in a steep gully located in the center of the site, requiring diversion of flood flows. | An upstream diversion channel was constructed to convey flows around the cell into a different drainage basin. |
| Atlas Title II Site | The disposal cell is immediately adjacent to the Colorado River, with a drainage area of thousands of square miles. There is a potential for the river to erode and migrate towards the cell, where high river velocities could impinge on the cell. | The licensee provided a design to resist the erosional forces associated with river migration, proposing to construct a large rock apron in accordance with the suggested criteria of NUREG-1623. (It should be noted that a decision was later made to completely move the entire tailings pile.) |

A large amount of experience was gained at various sites where erosion was found to be a significant problem. The staff analyzed and became familiar with various design solutions to mitigate significant erosion problems. Based on this experience, the staff considers that the erosion protection criteria suggested in NUREG-1623, combined with the graded regulatory approach discussed in this example, will provide a significant degree of flexibility in solving difficult erosion problems at complex sites.

*P.2.8.4 NRC Staff Inspections and Reviews of Construction Deficiencies*

For the last 15–20 years, NRC staff has routinely inspected sites in the Title I and Title II programs. The inspections included evaluations of soil cover and rock cover placement during construction and final closeout inspections of the completed work.

During these inspections, the staff noted that adequate placement of rock riprap layers was difficult for many contractors to achieve. There were numerous instances where rock layers were not placed to correct and consistent design thicknesses, in-place riprap layers did not have correct gradations and varied from tested samples, and areas of segregation where rock sizes were much smaller than required over large areas of the disposal cell. In many cases, the staff required corrective actions. Based on these problems encountered in staff reviews and its

experience with corrective actions, the staff developed guidance for proper rock placement. The staff used its construction experience to develop suggested quality control procedures for use by contractors. Appendix D to NUREG-1623 contains the suggested guidance.

At the present time, the staff also routinely accompanies DOE and/or DOE contractors on annual inspections at some sites where DOE now has custody and licensed responsibilities. These inspections have generally shown that covers are performing rather well and regulatory requirements continue to be met in all cases.

The staff also has experience with rock durability problems at uranium mill tailings reclamation sites. These problems occurred, for example, at sites using rock sources consisting of minerals susceptible to weathering (such as olivine basalts) or sources that had undergone hydrothermal alteration. In addition, the staff reviewed information on 149 case histories (E.E. Esmiol, ("Rock as Upstream Slope Protection for Earth Dams—149 Case Histories," issued September 1967 (Esmiol, 1967)), associated with rock durability problems at facilities constructed by other Federal agencies (such as Bureau of Reclamation dam sites). The NRC developed the rock durability criteria suggested in NUREG-1623 using the lessons learned from both uranium mill sites and other Federal sites.

### *P.2.8.5 Use of Design Bases Appropriate for Long Stability Periods*

As discussed in previous sections, erosion protection designs that meet the suggested criteria and guidance in this document and NUREG-1623 will provide adequate protection against extreme erosion events that could occur over the period of regulatory interest. The guidance reflects staff review and construction experience, past practices with regard to the selection of design bases, and good engineering practices. The staff considers that it has provided appropriate guidance on (1) selection of conservative rainfall and flooding events that reflect the long stability periods needed to meet regulatory requirements, (2) selection of parameters for determining flood discharges that account for the uncertainties associated with flood calculations, (3) computation of flood discharges using appropriate and/or conservative methods, (4) computation of appropriate flood levels and flood forces associated with the design discharge, (5) use of widely accepted, state-of-the-art, and standardized methods for determining erosion protection sizes and thicknesses, (6) selection of a rock type for the riprap layer that will be durable and maintain its size and ability to provide protection for a long period of time, and (7) placement of riprap layers in accordance with accepted engineering practice and in accordance with appropriate testing and QA controls.

### *P.2.8.6 Archaeological and Natural Bases for Long-Term Stability*

A strong archaeological basis exists to demonstrate the long-term stability of erosion protection materials. NUREG/CR-2642 presents substantial information to demonstrate the long-term survivability of various rock structures. NUREG-1623 provides information on long-term weathering rates, based on observations of rock petroglyphs that could be dated to a period of nearly 1,000 years before present. Further, as discussed in Section P.1.3 of this document, observations of Native American burial mounds in West Virginia, Ohio, Illinois, and Louisiana illustrate the survivability of man-made earthen structures for long periods of time (1,000 to 5,500 years).

Numerous other archaeological sites, such as Stonehenge (constructed about 4,000 years ago of diabase and silica cemented sandstone), Hadrian's Wall (constructed by the Romans over 2,000 years ago of primarily diabase), and numerous buildings in Europe demonstrate that

certain rock types used in these structures have been resistant to weathering over time periods greater than the 1,000-year period of regulatory compliance. In addition to archeological and historical evidence, natural rock sites can offer further insights that certain rock types and sources have been resistant to weathering. For example, dating of glacial erratics (rocks transported by glaciers) and preservation of river scour features in diabase results from high flow during glacial melting demonstrate that these rocks have been resistant to weathering and preserved for over 10,000 years.

The natural and archaeological insights noted above serve to demonstrate that engineered structures and construction materials can survive for very long periods of time. The intent of this guidance is to develop procedures for improving long-term stability by further enhancing design concepts where structures have generally remained intact for many years.

### P.2.9 Degradation Mechanisms for Erosion Control Covers

The erosion control cover at a typical decommissioning site could consist of either a rock layer or a soil layer and underlying rock layer. One of the most likely degradation mechanisms would be gully erosion. To account for that process, the design consideration that should be analyzed (and is considered by the staff to be the most likely) is the formation of a gully in the top soil cover, caused by surface erosion, flow concentrations, and/or the uprooting of large trees. The erosion should be assumed to continue and be deep enough to expose the rock layer, and thus the rock layer would need to be designed to resist further erosion and down-cutting of the gully. The licensee should design the soil cover, as described above, to be stable for rainfall events and runoff as large as the PMP/PMF. Further, the rock layer should be designed as a separate backup system for the soil cover and should also be designed for the PMP/PMF occurring in that gully, with flow concentrations produced by the growth of a drainage network to the gully. Further, the rock should meet the durability criteria suggested in NUREG-1623, with particular emphasis placed on the petrographic examination that indicates that no clay minerals are present (see Table P.4).

## P.3 Summary of Existing Guidance and Reference Information

Table P.4 summarizes existing guidance and reference information that may have some relevance to the application of engineered barriers at decommissioning sites. Early contact with the NRC staff is encouraged to discuss which portions of these referenced reports may be appropriate for the site and for the intended purpose of the engineered barriers. NUREG-1620, "Standard Review Plan for the Review of a Reclamation Plan for Mill Tailings Sites Under Title II of the Uranium Mill Tailings Radiation Control Act, Draft," Revision 1, issued June 2003, and NUREG-1623 contain guidance on the design of engineered disposal cells for uranium mill tailings sites. For sites considering engineered disposal cells for long-term stability (e.g., erosion control), this guidance may be somewhat useful. However, the standards in 10 CFR Part 40, Appendix A, applicable to uranium mills, are more prescriptive than the performance-based dose criteria of 10 CFR Part 20, Subpart E. Licensees using the uranium mill guidance should also consider how the guidance can be adapted for applicability to compliance with 10 CFR Part 20, Subpart E. As discussed below, in some cases, traditional designs have not performed particularly well with respect to infiltration.

A variety of programs have evaluated and continue to evaluate engineered barrier technology for waste containment. A comprehensive summary of engineered barrier research is not attempted in this document. However, some good examples of programs to understand,

design, and support engineered barrier performance applicable to decommissioning sites include the following:

- the U.S. Environmental Protection Agency (EPA), through the Alternative Cover Assessment Program (ACAP), has supported the field-scale evaluation of engineered covers

- the U.S. Department of Energy (DOE), through the Alternative Landfill Cover Demonstration (ALCD) completed a large-scale field demonstration comparing six landfill cover designs

- DOE has instrumented engineered covers at some Uranium Mill Tailings Remedial Action (UMTRA) sites to understand and evaluate their performance

- the University of Wisconsin conducted follow-up tests using exhumed ACAP covers under an NRC contract with the U.S. Geological Survey (USGS). The findings of the research project are documented in NUREG/CR-7028

- NUREG/CP-0195 summarizes experience with engineered barriers and recommendations from Federal, State, and tribal agencies, and academic and industry experts

- summarizes results of studies of barriers at four in-service Uranium Mill Tailings Radiation Control Act (UMTRCA) sites, focusing on hydraulic conductivity and radon flux

EPA's ACAP evaluated 27 test covers at 12 sites in eight states to characterize the field hydrology of water balance and conventional covers (Albright et al., 2004, NUREG/CR-7028). The evaluation included 12 conventional covers (7 composite and 5 clay) and 15 water balance covers (9 monolithic and 6 capillary barriers). Nine of the sites had side-by-side comparisons of conventional and alternative covers. Large-scale lysimeters (approximately 10 m by 20 m (30 ft by 70 ft) areal extent) were installed and instrumented to collect detailed water balance information. The main lessons learned for water balance covers were that (1) percolation rates in semiarid and sub-humid climates can be very low (less than 1 mm/y), provided that there is adequate storage capacity and that the vegetation effectively removes stored water each year, (2) there is a need to better understand the phenology of plants and the response to meteorological and geotechnical conditions, and (3) low percolation rates may not be achieved at sites with water balance covers, in particular at humid sites, but the water balance covers may still provide some performance benefit. The main lessons learned for composite covers were that (1) composite covers may be effective at limiting percolation to less than 1 mm/y while the geomembranes or geosynthetics are intact, and (2) clay covers are prone to damage over very short periods of time and can transmit percolation at much higher rates than anticipated. The ACAP program provides an excellent example of developing the technical basis for engineered barrier performance.

DOE's ALCD evaluated six cover designs at Sandia National Laboratory in Albuquerque, NM, to obtain large-scale water balance field data subjected to identical field and climatic conditions (Dwyer, "Water Balance Measurements and Computer Simulations of Landfill Covers," issued 2003). The study included a Resource Conservation and Recovery Act (RCRA) Subtitle D cover, a geosynthetic clay layer cover, a RCRA Subtitle C cover, an anisotropic barrier cover, a capillary barrier cover, and an evapotranspiration cover. The RCRA Subtitle D cover had the highest percolation rate (above the 1 mm/y goal), and the geosynthetic clay layer cover had the

second highest average percolation rate. Various damage processes, such as desiccation cracking, led to preferential flow through the RCRA Subtitle D cover. The field data from this project were interpreted to suggest the geosynthetic clay layer cover performance was suspect in this application, potentially due to desiccation and ion exchange. The other four cover types all had average annual fluxes of less than 0.2 mm/y (0.01 in/y).

Disposal cell covers for uranium mill tailings wastes at UMTRCA sites have been developed over the past 20 years at a variety of sites with different climates. From an erosion control and stability perspective, these covers have required little to no maintenance to prevent erosional release of radioactive materials although there have been problems at some sites with localized erosion events, such as slumping and ponding of water on the cover. At some covers, DOE has removed vegetation; however, these actions have primarily been undertaken because of concerns with the impact of the vegetation on water management (e.g., infiltration). Recent work shows that the presence of large deep-rooted plants both on rip-rap and vegetated covers, can result in increased releases of radon due to drying of the radon barrier while nearby unvegetated control areas have low radon fluxes (NUREG/CP-0312). DOE has instrumented some of the covers to understand and evaluate their performance. In particular, some covers based on resistive-type designs (e.g., impermeable layers), similar to unlined RCRA Subtitle D covers, have not achieved the design values for hydraulic conductivity measured in the laboratory and therefore appear to have much higher infiltration rates than anticipated (Waugh, "Design, Performance, and Sustainability of Engineered Covers for Uranium Mill Tailings," issued 2004). In fact, at a number of sites, the in situ hydraulic conductivity was measured to be more than two orders of magnitude higher than the design target. However, monitoring data from evapotranspiration covers suggest that design infiltration rates have been achieved. One of the lessons learned was that seemingly subtle differences in soil types, sources, and compaction can result in significant differences in performance.

A number of the covers that were part of the ACAP study were exhumed and studied following termination of the program. The University of Wisconsin conducted field and laboratory tests under an NRC contract with USGS to study the change in engineered properties of the cover materials during their 5- to 10-year service life. NUREG/CR-7028 documents the findings of the research project. An important conclusion of the report is that compacted soil materials used in cover materials at the sites studied did not retain "as built" properties over periods of regulatory interest. The properties of these materials change to values more typical of surrounding soils within 5 to 10 years after installation. Changes in low permeable cover soils can be rapid and can result in an increase to the saturated hydraulic conductivity by three to four orders of magnitude. With respect to drainage layers, greater reductions in transmissivity and permeability were observed for drainage layers covered with soils having higher fines content. However, this effect was modest, and all of the drainage layers functioned as anticipated. It is important to note that the report did not address cover elements designed for erosion protection.

With regard to covers that use composite materials, NUREG/CR-7028 found that these types of covers do not appear to significantly degrade over the short term. An analysis in the report showed that GCLs have very low saturated hydraulic conductivity (less than $5 \times 10^{-11}$ m/s) when placed on a moist subgrade (water content greater than 10 percent) and covered with a geomembrane and cover soil soon after installation, although GCLs installed under other conditions can be much more permeable. In addition, changes in geomembranes and geosynthetic drainage layers during the short period of the study were modest or small. GCLs that underwent complete hydration maintained low hydraulic conductivity, even when the native sodium (Na) was replaced by calcium (Ca) and magnesium (Mg). However, GCLs installed under other conditions were much more permeable; therefore, QA during cover construction is

of great importance. Analysis of antioxidants in the geomembranes showed that antioxidant depletion was consistent with expectations based on first-order kinetics and laboratory-measured depletion rates. Based on these rates, the minimum service life of geomembranes was estimated to be on the order of 50–125 years. The report concluded that actual service lives are likely to be longer but are difficult to estimate. Because covers change over time, it was recommended that they be monitored to ensure that they are functioning as intended. Recommendations included monitoring using large pan lysimeters, combined with secondary measurements collected for interpretive purposes (e.g., water content, temperature, vegetation surveys), as well as studying analogs of natural environments.

Given the risk-significant findings of NUREG/CR-7028, the NRC staff review assessed the risk associated with various licensed facilities, including a single decommissioning site with an operating cover. The risks associated with underperformance of an engineered cover is related to a variety of factors, including climate, evapotranspiration rates, hydrogeological properties of vadose zone materials, depth to groundwater, thickness of overburden materials, depth of resistive barriers (with shallower covers potentially more susceptible to surface degradation processes), existence of a liner and leachate collection system, and the extent to which groundwater is used as a resource. The NRC staff recommendations focus on uranium recovery facilities and include continued monitoring; data collection; stakeholder communication; and research, including field studies to continue to evaluate degradation mechanisms and to study the uncertainty associated with long-term performance of engineered covers.

In August 2010, the NRC sponsored a workshop at its headquarters office in Rockville, MD (NUREG/CP-0195). The purpose was to provide Federal, State, and tribal agencies, and other interested stakeholders a forum to share information on engineered surface covers and bottom liners used for waste containment at low-level waste (LLW) disposal facilities, decommissioning sites, and uranium mill tailings sites. Stated objectives of the workshop included (1) facilitation of communications among Federal and State staff and contractors and selected experts on current engineered barrier issues and technical and regulatory experiences, (2) discussion of lessons learned and approaches for monitoring and modeling, (3) preparation of recommendations to address maintenance of engineered barrier performance over time, and (4) identification of topics for future research and the potential need to update technical guidance. At the conclusion of the workshop, it was noted that consistent themes repeatedly mentioned by the workshop participants were (1) the need to more actively monitor and quantify engineered system behavior so as to know the record of performance and better understand the key processes effecting performance, and (2) the need for better communication and exchange between the specialists of various fields, (e.g., engineers, pedologists, ecologists, modelers) involved with the design, construction, and maintenance of engineered barriers. These insights were supported by technical presentation details and discussions by all engaged participants and were repeated throughout the 3-day workshop.

Following the August 2010 workshop additional studies were conducted to address workshop recommendations. For example, NUREG/CR-7200 investigates the coupling of erosion and hydrology and its impact on the performance of waste covers. Hydrologic and landform evolution models were used to evaluate various engineered cover designs for LLW disposal with respect to the ability of the cover designs to limit erosion and percolation of water through the waste. Climate and vegetation were found to have the most influence on erosion with erosion depths in semi-arid climates being 4 m greater compared to humid climates for simulations with a rip-rap or gravel admixture surface. Vegetation was also found to decrease erosion and percolation through the cover. Short slope, low grade, and small grade differences

(for semi-arid climates) or terraced (for humid climates) slopes were found to be optimal with respect to limiting erosion. Although rip-rap surface materials decreased erosion, use of rip-rap led to higher percolation through the cover. In contrast, a gravel admixture surface had slightly greater erosion but was much were effective in limiting percolation through the cover.

A workshop was held at NRC headquarters on July 25-26, 2018, to present and discuss findings from a research project in which the performance of radon barriers at UMTRCA sites was studied after having been in service for about 20 years. Four mill tailing sites were visited by the research team: Falls City in Texas, Bluewater in New Mexico, Shirley Basin South in Wyoming, and Lakeview in Oregon. Small areas on these sites were excavated, radon fluxes were measured, numerous observations were made, and samples taken for a variety of parameters such as root counts, moisture, density, Pb-210 concentrations, hydraulic conductivity, soil texture, chemistry, and nematode counts. As part of the project, a comparison of the current state of these engineered covers to their as-built condition was made. In some cases, the saturated hydraulic conductivity and radon flux of the covers increased over time to higher values closer to values observed in natural analogs. In one case, the saturated hydraulic conductivity of the cover remained fairly stable over the time period of the study (approximately 20 years), likely due to the high percentage of expandable clays in the cover material. Biotic activity, especially in the form of root intrusion into the radon barrier, was found to have a significant deleterious impact on radon barrier performance. The conference proceedings NUREG/CP-0312 contains extended abstracts and slides from the presentations, and a summary of the results and recommendations from the workshop.

The programs cited above, and the documents listed below are not intended to be comprehensive; rather, they are intended to provide appropriate examples of studies undertaken to understand and support engineered barrier performance. The examples provided above focused on engineered covers, which are only one type of barrier addressed in this guidance. The documents listed below provide information on a variety of different barriers.

**Table P.4 Summary of Existing Key Documents Related to Engineered Barriers**

| Document | Brief Summary |
|---|---|
| Walton, J.C., L.E. Plansky, and R.W. Smith, "Models for Estimation of Service Life of Concrete Barriers in Low-Level Radioactive Waste Disposal," NUREG/CR-5542, U.S. Nuclear Regulatory Commission, Washington, DC, September 1990. | Provides primarily empirically based models for typical concrete formulations to estimate degradation rates. |
| Bennett, R.D., "Recommendations to the NRC for Soil Cover Systems Over Uranium Mill Tailings and Low-Level Radioactive Wastes—Identification and Ranking of Soils for Disposal Facility Covers," NUREG/CR-5432, U.S. Nuclear Regulatory Commission, Washington, DC, February 1991. | Discusses (1) selecting soil materials, (2) laboratory and field tests for covers, and (3) construction methods. |
| NISTIR 5612, "4SIGHT, Manual: A Computer Program for Modeling Degradation of Underground LLW Concrete Vaults," National Institute of Standards and Technology, Gaithersburg, MD, 1995. | User manual for numerical computer modeling of concrete degradation, 4SIGHT, to facilitate assessment of concrete vaults for isolating radioactive waste in low-level waste disposal applications. |
| NISTIR 89-4086, NUREG/CR-5466, "Service Life of Concrete," National Institute of Standards and Technology, Gaithersburg, MD, 1995. | Examines degradation processes in cement-based materials and discusses considerations of their occurrence, extent of potential damage, and mechanisms. |
| NUREG-1532, "Final Technical Evaluation Report for the Proposed Revised Reclamation Plan for the Atlas Corporation Moab Mill," U.S. Nuclear Regulatory Commission, Washington, DC, March 1997. | Section 4 provides an example of the staff review of a reclamation design and discusses staff bases for acceptability of rock riprap erosion protection and input parameters used for those designs. |
| NISTIR 6519, "Effect of Drying Shrinkage Cracks and Flexural Cracks on Concrete Bulk Permeability," National Institute of Standards and Technology, Gaithersburg, MD, 2000. | Discusses a model for predicting both the width and spacing of flexural and drying-shrinkage cracks to estimate composite (intact and cracked) concrete structure permeability. |
| U.S. NRC, "A Performance Assessment Methodology for Low-Level Radioactive Waste Disposal Facilities, Recommendations of NRC's Performance Assessment Working Group," NUREG-1573, U.S. Nuclear Regulatory Commission, Washington, DC, October 2000. | Provides general information pertinent to modeling and assessment of engineered barriers. Provides a bibliography of reports related to engineered barriers. |
| NISTIR 6747, "Validation and Modification of the 4SIGHT Computer Program," National Institute of Standards and Technology, Gaithersburg, MD, 2001. | Discusses the validation and verification of the fluid transport mechanisms incorporated in the concrete degradation code 4SIGHT using reference and laboratory data. |

**Table P.4 Summary of Existing Key Documents Related to Engineered Barriers (cont.)**

| Document | Brief Summary |
|---|---|
| EPA/600/R-02/099, "Assessment and Recommendations for Improving the Performance of Waste Containment Systems," U.S. Environmental Protection Agency, 2002. | Discusses issues related to the design, construction, and performance of waste containment systems used in landfills, surface impoundments, and waste piles and in the remediation of contaminated sites. |
| Johnson, T.L., "Design of Erosion Protection for Long-Term Stabilization," NUREG-1623, U.S. Nuclear Regulatory Commission, Washington, DC, September 2002. | Provides guidance on methods to achieve erosion controls for long-term stabilization. Provides a list of key references, including the technical work supporting the guidance. |
| Dwyer, Stephen F., "Water Balance Measurements and Computer Simulations of Landfill Covers," PhD Dissertation, University of New Mexico, 2003. | Provides a comprehensive summary of data collection, analysis, and computer simulations associated with DOE's ALCD program. Also includes a summary of measurements of infiltration at various sites with engineered covers. |
| "Handbook of Groundwater Remediation Using Permeable Reactive Barriers," edited by D.L. Naftz, S.J. Morrison, C. Fuller, and J.A. Davis, 2003. | Discusses cost-effective remedies and provides case studies on treating contaminated groundwater using permeable reactive barriers, including discussions on construction, development of reactive materials, and operable chemical and biological reactions. |
| Interstate Technology & Regulatory Council (ITRC), "Technical and Regulatory Guidance for Design, Installation, and Monitoring of Alternative Final Landfill Covers," Washington, DC, 2003. | A guidance document primarily written for decisionmakers associated with plan development, review, and implementation of alternative covers, focuses on the decisions and facilitating the decision processes related to the design, evaluation, construction, and postclosure care associated with alternative covers. |
| NAS, "Assessment of the Performance of Engineered Waste Containment Barriers," National Academy of Sciences, Washington, D.C., 2007. | Identifies engineered barrier systems used for surface and subsurface waste containment, defines and evaluates performance of barriers, assesses methodologies to predict and monitor performance, and identifies information needs to fill knowledge gaps. The report provides a series of recommendations on data collection and distribution; models; monitoring periods, and performance criteria. |
| NISTIR 7026, "Condition Assessment of Concrete Nuclear Structures Considered for Entombment," National Institute of Standards and Technology, Gaithersburg, MD, 2003. | Provides assessment of cement-based engineered barrier structures based on characterization of intact concrete and crack properties. Material property uncertainties are incorporated into a Monte Carlo simulation. |

**Table P.4 Summary of Existing Key Documents Related to Engineered Barriers (cont.)**

| Document | Brief Summary |
|---|---|
| Albright, W.H., et al., "Field Water Balance of Landfill Final Covers," Journal of Environmental Quality, 33(6), 2317–2332, 2004. | Results of large-scale field research study to assess the ability of landfill final covers to control infiltration into underlying waste. A comprehensive publication summarizing ACAP experience. |
| Waugh, W.J., "Design, Performance, and Sustainability of Engineered Covers for Uranium Mill Tailings," Proceedings of Long-Term Performance Monitoring of Metals and Radionuclides in the Subsurface: Strategies, Tools, and Case Studies. U.S. Environmental Protection Agency, U.S. Department of Energy, U.S. Geological Survey, U.S. Nuclear Regulatory Commission, April 21–22, 2004, Reston, VA, 2004. | Provides information on experiences with cover designs for DOE's UMTRA Project sites of conventional and alternative covers. |
| Interstate Technology & Regulatory Council, "Permeable Reactive Barriers: Lessons Learned/New Directions," Washington, DC, 2005. | Summarizes the understanding and experience with permeable reactive barriers, including numerous case studies. |
| "Long-Term Performance of Permeable Reactive Barriers, edited by K.E. Roehl, T. Meggyes, F.G. Simon, and D.I. Stewart, 2005. | Describes methods for evaluating and enhancing the long-term performance of permeable reactive barrier systems, focused on systems that treat uranium, and organic contaminants by sorption and/or precipitation mechanisms and discusses a number of topics, including (1) selection, characterization, and development of suitable reactive materials, (2) identification of attenuation mechanisms, (3) accelerated testing, (4) evaluation of the influence of site conditions, and (5) monitoring techniques. |
| National Research Council, National Academy of Sciences, "Assessment of the Performance of Engineered Waste Containment Barriers," 2007. | Examines current knowledge and gaps in the understanding of the performance of engineered waste containment barriers and recommends research needs in the area. |
| Walter, G.R., P. Dubreuilh, "Evaluation of Approaches to Simulate Engineered Cover Performance and Degradation," Prepared for U.S. NRC Contract NRC-02-07-006, Center for Nuclear Waste Regulatory Analyses, San Antonio, TX, April 2007. | Reviews a report of 21 codes, consisting of (1) hydrologic codes for evaluating percolation through the cover, (2) generalized and localized erosion codes for evaluating long-term stability of the cover, and (3) miscellaneous codes for evaluating degradation of covers. |

**Table P.4 Summary of Existing Key Documents Related to Engineered Barriers (cont.)**

| Document | Brief Summary |
|---|---|
| Albright, W.H., Benson, C.H., Waugh, W.J., "Water Balance Covers for Waste Containment: Principles and Practice," American Society of Civil Engineers, Reston, VA, 2010. | Textbook on evapotranspiration covers using information from numerous studies examining performance and degradation mechanisms of various engineered surface barrier components over time. |
| Arlt, H., "A Regulatory Perspective of Monitoring and Assessing Performance of Engineered Surface Barriers," Proceedings of Waste Management Symposium, Phoenix, AZ, 2010. | Paper discussing function and type of engineered surface barriers, as well as approaches to monitoring and assessing time-dependent performance and degradation from a regulatory perspective. |
| Benson, C.H., et al., "Engineered Covers for Waste Containment: Changes in Engineering Properties & Implications for Long-Term Performance Assessment," NUREG/CR-7028, U.S. Nuclear Regulatory Commission, Washington, DC, December 2011. | Examines changes in properties of engineered covers used for waste containment and discusses practical implications on long-term performance assessment, as well as providing recommendations. |
| Arlt, Hans, R. L. Johnson, D. Mandeville, G. Alexander, M. Meyer, J. Philip, J. Kanney, M. Fuhrmann, and T. Johnson, 2011, *Documentation of the Engineered Covers Technical Group (ECTG) Activities. ML112300105* | A qualitative assessment of existing sites to identify and prioritize by risk existing and planned covers that may be impacted by the findings in NUREG/CR-7028. Processes contributing to risk include both radon flux from the disposal cell and effects to the groundwater due to the potential of an increased rate of water infiltration through the covers. |
| Interstate Technology and Regulatory Council, "Permeable Reactive Barrier: Technology Update Version 1.01," 2011. | Provides consolidated information from previously issued ITRC documentation on additional types of reactive media, contaminants amenable to treatment, longevity of barriers, and approaches to construction and installation of barriers. |
| Nicholson, T.J., and H.D. Arlt, "Proceedings of the Workshop on Engineered Barrier Performance Related to Low-Level Radioactive Waste, Decommissioning, and Uranium Mill Tailings Facilities: Held at the U.S. Nuclear Regulatory Commission Headquarters, Rockville, MD, August 3-5, 2010," NUREG/CP-0195, U.S. Nuclear Regulatory Commission, Washington, DC, August 2011. | Summarizes experience with engineered barriers and recommendations from Federal and State agencies, Tribal Nations, and academic and industry experts compiled during engineered barrier workshop at the NRC. |

**Table P.4 Summary of Existing Key Documents Related to Engineered Barriers (cont.)**

| Document | Brief Summary |
|---|---|
| Dinwiddie, C., S. Stothoff, "Methods for Monitoring Net Inflow Through Soil Covers of Uranium Mill Tailings Impoundments," Prepared for U.S. Nuclear Regulatory Commission by Center for Nuclear Waste Regulatory Analyses, San Antonio, Texas, November 2013. | The report examines approaches for implementing effective postclosure monitoring programs for estimating inflow through uranium mill tailings impoundment covers, focusing on U.S. Nuclear Regulatory Commission (NRC) licensed Title-II-in-closure sites. |
| Snyder, K.A., and W.J. Weiss, "Sensors and Monitoring to Assess Grout and Vault Behavior for Performance Assessment," NUREG/CR-7169, U.S. Nuclear Regulatory Commission, Washington, DC, June 2014. | Preliminarily evaluates state-of-the-art of sensors, nondestructive evaluation methods, and any relevant geophysical techniques that may be used to quantify changes to the intended chemical (e.g., redox state) and structural properties (e.g., crack initiation, development, and propagation) of large engineered waste isolation systems. |
| Sagar, B., et al., "Uranium Tailings Impoundments—Bottom Liners and Their Performance," Prepared for U.S. Nuclear Regulatory Commission by Center for Nuclear Waste Regulatory Analyses, San Antonio, Texas, January 2015. | The focus of the report is on providing information to address some uncertainties associated with the use of bottom liners under tailings impoundments at uranium recovery facilities. In addition, the information can be used to provide insights regarding the long-term performance expectations of bottom liners at tailings impoundments. |
| Smith, C.L., C.H. Benson, "Influence of Coupling Erosion and Hydrology on the Long-Term Performance of Engineered Surface Barriers," NUREG/CR-7200, U.S. Nuclear Regulatory Commission, Washington, D.C., May 2016. | Hydrologic and landform evolution models were used to evaluate various engineered cover designs for LLW disposal facilities with respect to the ability of the cover designs to limit erosion and percolation rates. |
| Fuhrmann, M., C.H. Benson, J. Waugh, M. Williams, and H. Arlt, "Proceedings of the Radon Barriers Workshop," NUREG/CP-0312, NRC: Washington, DC. July 25-26, 2018. | NUREG/CP-0312 summarizes results of studies of barriers at four in-service UMTRCA sites, focusing on hydraulic conductivity and radon flux. |

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Low-Level Radioactive Waste, Decommissioning, and Uranium Mill Tailings Facilities," NUREG/CP-0195. NRC: Washington, DC. 2011b.

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Pabalan, R.T., et al., "Review of Literature and Assessment of Factors Relevent to Performance of Grouted Systems for Radioactive Waste Disposal," CNWRA 2009-001, Prepared for Nuclear Regulatory Commission Contract NRC NRC-02-07-006. Center for Nuclear Waste Regulatory Analyses: San Antonio, Texas. 2009.

Smith, C.L., C.H. Benson. "Influence of Coupling Erosion and Hydrology on the Long-Term Performance of Engineered Surface Barriers," NUREG/CR-7200, U.S. Nuclear Regulatory Commission: Washington, DC. 2016.

Waugh, W.J., et al. "Ecology, Design, and Long-Term Performance of Surface Barriers: Applications at a Uranium Mill Tailings Site," Presented in "Barrier Technologies for Environmental Management," Summary of a Workshop, National Research Council, National Academy of Sciences. 1997.

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# APPENDIX Q

# UNCERTAINTY IN PERFORMANCE AND DOSE ASSESSMENTS

## Q.1 Introduction

The objective of this appendix is to provide information and guidance to the NRC staff on the treatment and representation of uncertainty in site-specific dose or performance assessments, DPs, and technical analyses. This guidance discusses types of uncertainty, methods to understand the impact of uncertainty, and different approaches to incorporate uncertainty in these assessments. The appendix provides a generic example of the results of different methods to represent uncertainty in dose and performance assessments.

## Q.2 Background

A performance assessment is a projection of what can happen, how likely it is, and the associated consequences for potentially many thousands of years into the future. Dose assessments typically incorporate uncertainty and variability in data and parameters. Although dose or performance assessments may be probabilistic or deterministic, modern computational advances have allowed greater use of probabilistic assessments. Dose and performance assessments can be data intensive, and in many cases, data can be sparse. Sparse site-specific data typically lead to the use of generic sources of information, where the basis may range from limited to fairly complete data sets. If caution is not taken, misinterpretation or misrepresentation of the variance in information, particularly when site-specific supporting information is sparse, may result in ineffective decision-making. Therefore, results produced from models with sparse data and limited site-specific information can require careful interpretation.

Figure Q.1 provides a conceptual overview of the dose and performance assessment process. The central portion of the figure has four main steps: collect data, develop conceptual models, develop numerical and computer models, and combine models and estimate effects. Each step in the process can be influenced by uncertainty in different ways. Usually consideration of uncertainty in dose and performance assessments is focused on data uncertainty and, in particular, a subset of overall data uncertainty (i.e., observed data uncertainty). This appendix covers data uncertainty, as well as uncertainty about process and conceptual models (model uncertainty) and model representation or computational uncertainty.

It is important to consider uncertainty in dose and performance assessments, because the assessments are designed to provide estimates of future system performance for consideration against regulatory criteria. If uncertainty has not been appropriately represented in the results of the assessments, regulatory decision-making may be impaired or inappropriately biased. This appendix is organized into the following main sections:

- Section Q.2 of this appendix provides examples of different types of uncertainty that may be encountered in a dose or performance assessment.
- Section Q.3 discusses high-level topics and complications that can arise as a result of incorporating uncertainty in a dose or performance assessment.
- Section Q.4 provides general questions that a reviewer should consider when evaluating uncertainty in a dose or performance assessment.
- Section Q.5 discusses different methods for evaluating the impact of different uncertainties on dose and performance assessment results.

- Section Q.6 briefly discusses different uncertainty and sensitivity analysis methods.

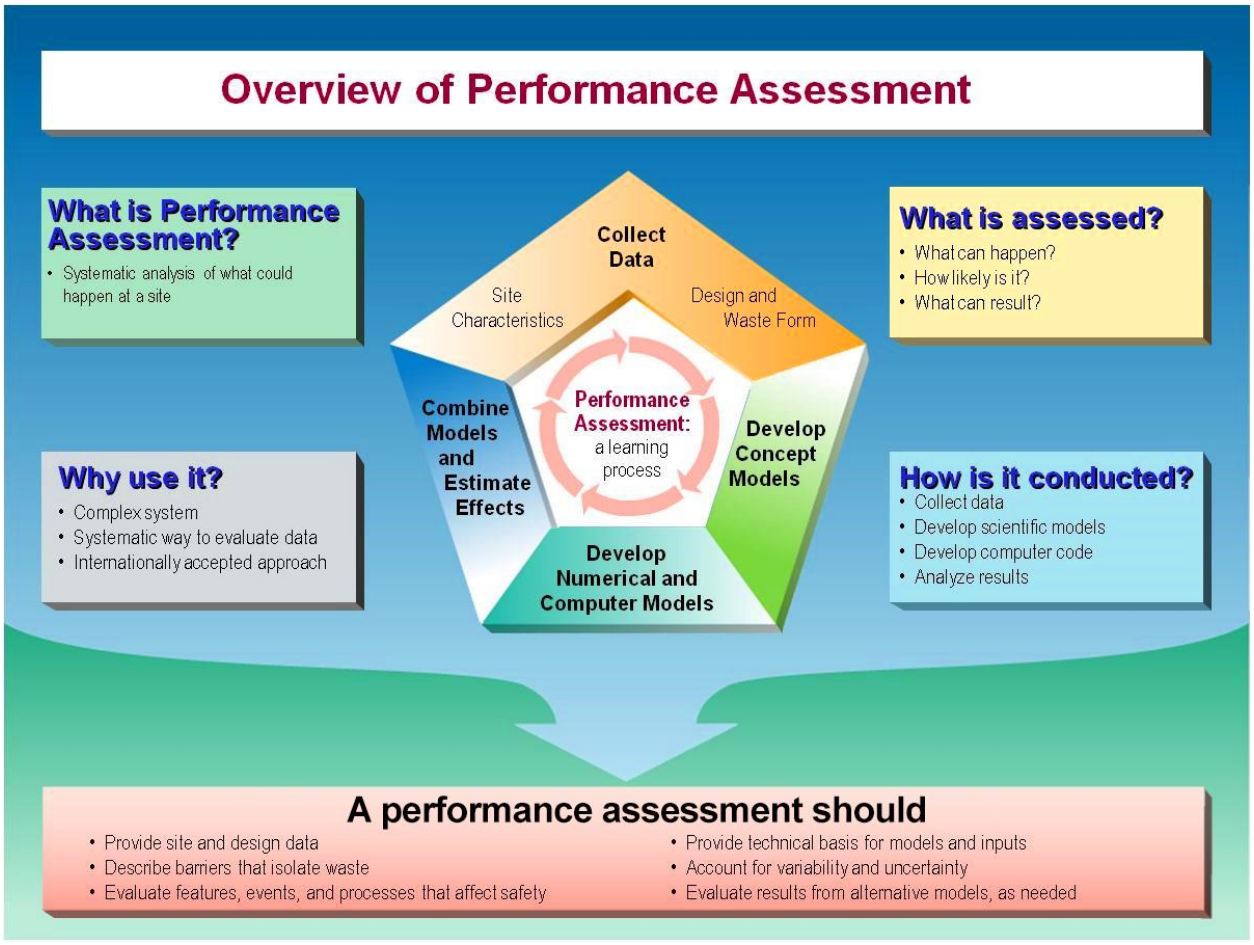


**Figure Q.1 Conceptual Overview of the Performance Assessment Process**

## Q.3 Types of Uncertainty

A variety of different sources of uncertainty can affect dose and performance assessment modeling. Figure Q.2 is a conceptual representation of the dose and performance assessment process and significant sources of uncertainty associated with the different steps in the process. At the fundamental level, uncertainty can be classified as epistemic or aleatory. Epistemic uncertainty is sometimes referred to as reducible uncertainty or lack of knowledge uncertainty. In theory, this type of uncertainty can be reduced by investing resources to develop understanding. In practice, reduction in epistemic uncertainty may not be warranted (i.e., not risk significant) or may be too expensive. An example of epistemic uncertainty is the porosity of concrete in a waste disposal vault. Aleatory uncertainty arises from a perceived randomness in the behavior of the system under study. Aleatory uncertainty involves the characterization of the likelihood of events that have a real possibility of occurrence due to fundamental natural processes. The appropriate separation of aleatory uncertainty and epistemic uncertainty can be an important part of the conceptual structure and associated computational analysis of a complex system. However, aside from the most complex PAs the staff may review, a separation of uncertainty is rarely made. In some cases, uncertainties cannot be readily classified as either epistemic or aleatory. A process that results in an observation that is perceived as random may not be understood, such that the effect of the process is classified as aleatory whereas ultimately it is the process that is epistemic.

Figure Q.2 is a conceptual figure to communicate the major sources of uncertainty in the dose and performance assessment process. Most PAs explicitly account for data uncertainty and will implicitly account for model uncertainty. Few explicitly or implicitly account for some subtypes of representational uncertainty, although some projects with rigorous QA programs may have requirements that broadly address representational uncertainty. There are also subtypes of uncertainty (e.g., integration uncertainty) within broader classes (model uncertainty).

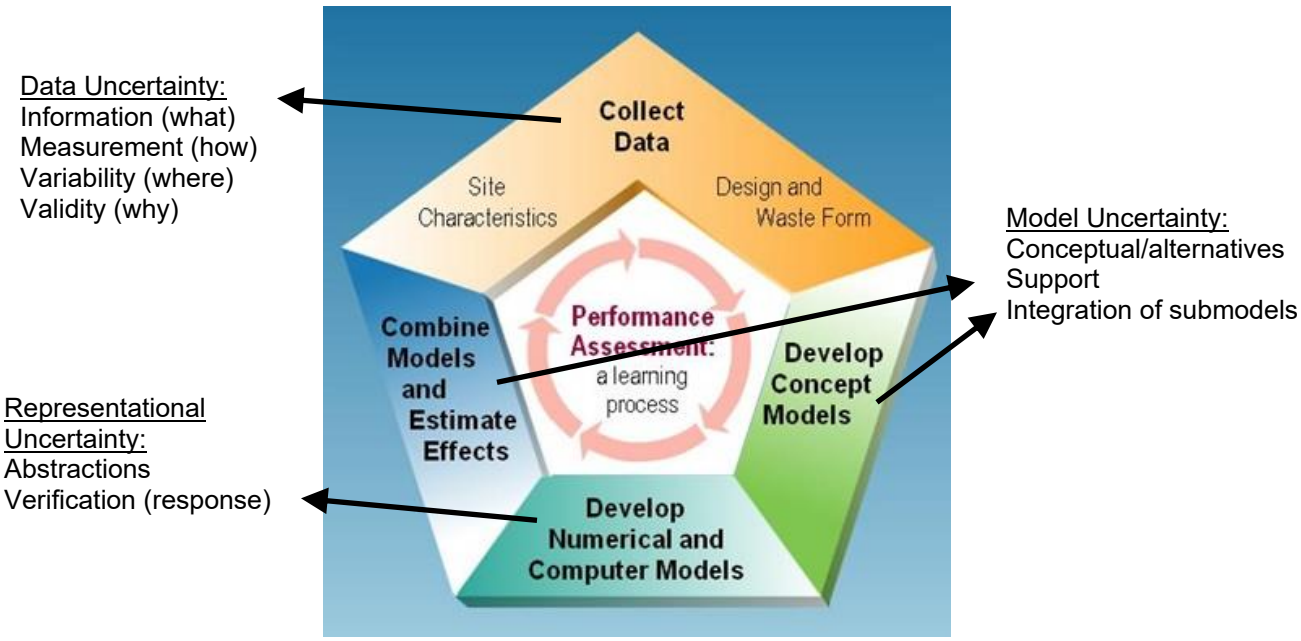


**Figure Q.2 Uncertainty Types Associated with Performance Assessment**

The following definitions are applied in this appendix:

*Uncertainty*—Uncertainty is a multifaceted characterization about data or predictions made from data that may include several concepts, including error, accuracy, validity, quality, noise, and confidence.

*Aleatory uncertainty*—Irreducible uncertainties stemming from an inherent randomness in processes.

*Epistemic uncertainty*—Reducible uncertainty or lack of knowledge uncertainty.

*Performance assessment model*—The integrated model of numerous submodels used to simulate the range of future performance.

*Submodel*—A component of a dose or performance assessment model (e.g., wasteform model, groundwater transport model) that may be an abstraction of a detailed process model.

*Abstraction*—A simplification of a process model that represents the essential behavior of the process model (e.g., lookup tables, response surfaces, and parameter distributions).

*Process model*—A detailed “physics-based” model used to simulate some aspect of the dose or performance assessment model, usually involving a solution of some form of differential equations, with a high (relative to a dose or performance assessment model) degree of spatial and temporal resolution, noting that computational resources may be high for using a process model.

*Verification*—The process of ensuring that the calculated values are correct, noting that a standard QA program includes verification of all technical aspects of the dose and performance assessment process.

*Model support*—Multiple lines of evidence used to support the results of a dose or performance assessment model, including but not limited to laboratory experiments, field experiments, field observations, natural analogs, expert elicitation, and comparison to alternative models.

## Q.4 General Considerations

The licensee should evaluate data, model, and representational uncertainty as part of the dose and performance assessment process. In general, a systematic and proactive approach to managing uncertainties is better than an ad hoc and reactive approach. In particular, for dose and performance assessment models, uncertainties may not be well understood. Limited information, as well as inherent complexity, can impair the development of understanding and appropriate management techniques.

Deterministic modeling is sometimes used for PAs. Typically, these types of PAs will attempt to use "conservative" parameters and models and evaluate uncertainties with one-at-a-time parameter sensitivity analyses. However, because dose and performance assessment models are collections of integrated submodels, local and subjective definitions of "conservative" may not be robust because of the inherent complexity in dose and performance assessment model response. In addition, one-at-a-time parameter sensitivity analyses are a local measure of the impact of an uncertainty. The term local is with respect to the dimensions of the uncertainty space. Deterministic modeling that can overcome these disadvantages can be an effective tool to provide information for decision-making.

As computational resources have improved, probabilistic modeling has become more common. Probabilistic modeling combined with global uncertainty analyses can provide more information with respect to the importance of different uncertainties and submodels compared to deterministic modeling. It can also be used to make more robust "conservative" modeling decisions. A primary drawback can occur when probabilistic modeling is used for a site-specific analysis with very limited site-specific information. Risk dilution occurs when favorable conditions (compared to the specific site) are included in parameter distributions and models that are not representative of the site-specific conditions.

The overall objective of dose and performance assessment modeling should be to make the models as simple as possible but no simpler. Stated another way, the complexity of the dose and performance assessment model should be consistent with available supporting information and be able to include the response to uncertainties. Even with advances in computational resources, complexity is not always warranted or desired. Adding complexity that does not address risk or advance decision-making impedes the review and understanding process and dilutes effort on the more significant aspects. On the other hand, a model without adequate complexity and model support may miss important processes or interactions. The dose and performance assessment process should be iterative, which allows the analyst to arrive at the appropriate level of complexity in an iterative manner. The complexity of a model will need to reflect a balance between needing to include essential behavior and avoiding the burden of nonessential behavior. If model support information is available, it can be used to help determine when the level of detail of the modeling is sufficient.

## Q.5 Review Procedures

This material is intended to be used in a risk-informed manner. For sites where the risk is confidently low or the analysis is clearly conservative, consideration of uncertainties is less important, and these procedures can be applied on a selective basis.

### Q.5.1 General Considerations

The following general considerations may be used to perform a high-level evaluation of the scope of the uncertainty assessment. These are written in question form to allow a reviewer to step through them and determine how the topics were addressed.

- Is the data (e.g., range, distribution type) representative of the site-specific conditions?
    - Does it vary spatially and temporally? If so, how has that variability been characterized in the data?
    - Can bounds to the data be established?
    - Has the data been measured? What is the measurement error? Is there bias in the measurements?
    - Did the data have outliers? How were they treated?
- Are correlations in the data needed?
- Have generic datasets been used?
    - Has the full range of the data been used?
    - Has risk dilution been introduced?
    - Is the generic dataset representative of site-specific conditions?
    - If a subset of the generic dataset has been used, how was it selected?
- Has adequate model support been provided or are there alternative conceptual models?
    - Have the alternative models been evaluated?
    - Have combinations of alterative models been evaluated?
    - Does the selected model have higher or lower risk?
    - Is the selected model most plausible? Why?
- Is there uncertainty associated with how submodels have been integrated?
    - How have the connections been established?
    - Is any of the information that is passed an observable metric?

- Are there feedbacks? If yes, how were they established?

- Were abstractions used in the modeling?
  - Does the abstraction adequately represent the process model?
  - What comparisons were made (e.g., visual, statistical)?
  - Has verification of the abstraction been performed?
  - Does the abstraction preserve important responses that would otherwise be present in the process model?

By addressing these high-level general questions, the reviewer should ensure that uncertainty has been appropriately handled in the modeling. Not all of these general considerations will apply in every review; however, in the majority of cases, these questions can be used as a form of checklist to evaluate the treatment of uncertainty.

## Q.5.2 Examples

The following are intended to provide illustrative examples of the technical topics contained in the previous section. The examples may be related to a specific discipline (e.g., geochemistry) but are to be used more generically by the reviewer. In other words, they are intended to provide practical examples of concepts.

### *Q.5.2.1 Data Representativeness*

Data must be representative of the site-specific conditions and be consistent with the observations. If not, use of data that are not representative could introduce errors into the dose and performance assessment.

The following example is for the liquid saturation in a layer of the unsaturated zone that is sufficiently below the surface to not be subject to diurnal variation. However, seasonal variation is significant. Figure Q.3 is a plot of 10 years of the liquid saturation of the layer in question showing the times that samples were collected. Over 100 samples were collected with a mean liquid saturation of 0.88 and a standard deviation of 0.11. Figure Q.4 is a contour plot of an areal view of the liquid saturation in a layer, showing the sample locations.

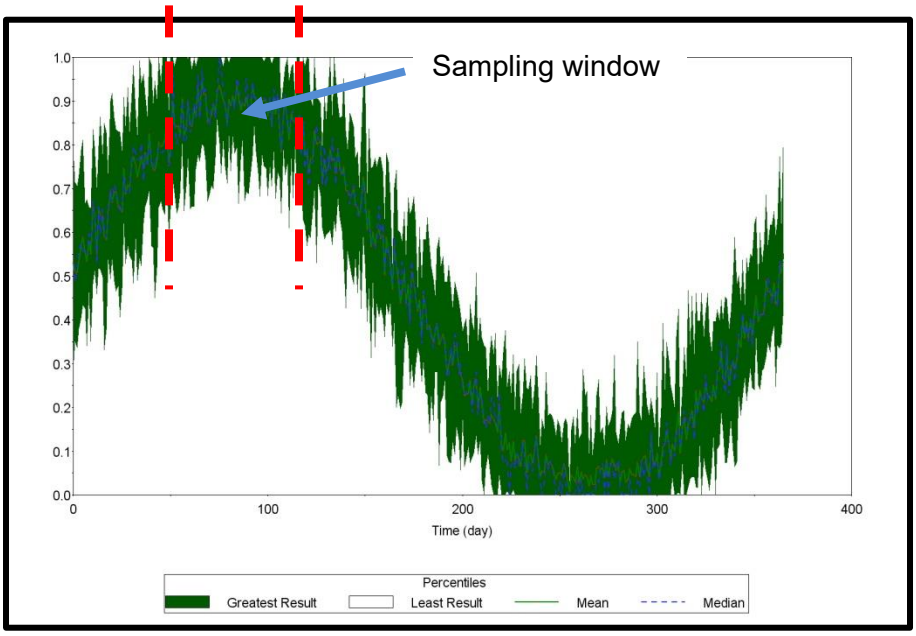


**Figure Q.3 Ten Years of Simulated Liquid Saturation Data**

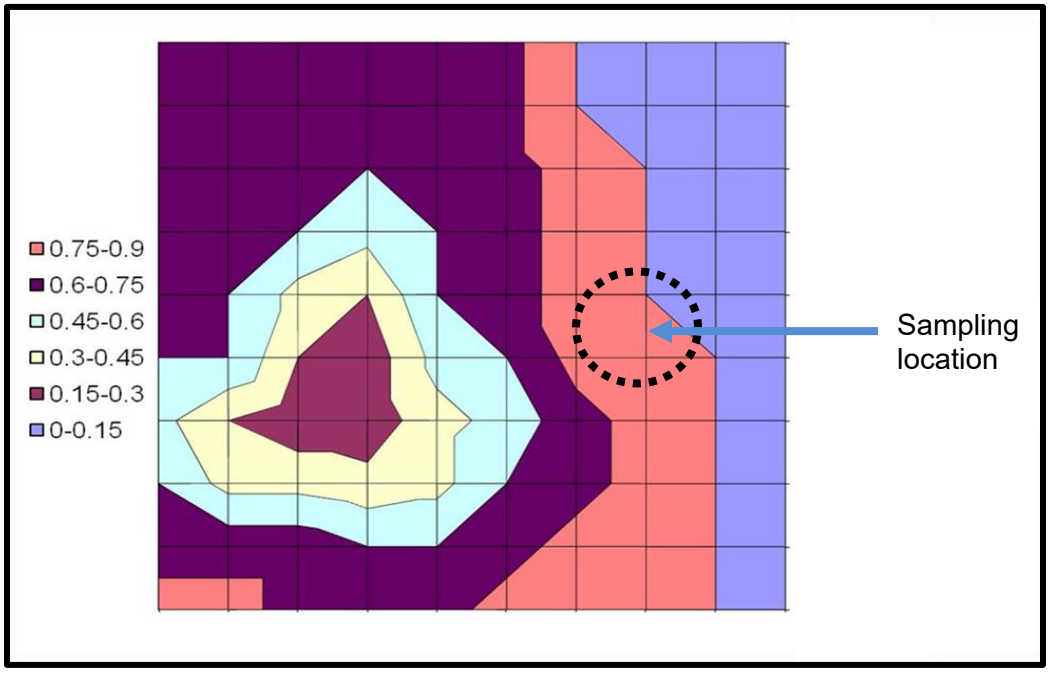


**Figure Q.4 Contour Plot of Liquid Saturation at a Given Point in Time**

Annual liquid saturation samples were collected in the early spring months each year, from fixed sampling locations. The example is intended to convey the importance of the representativeness of data. Even with many samples, if those samples do not characterize the temporal and spatial variability in the data, then simulations based on the data may be inaccurate. Temporal variability in environmental data may be driven by diurnal, seasonal, and longer-term fluctuations. Data used in a dose or performance assessment must adequately account for the different types of variability. Spatial variability may also play an important role. In the example shown in Figure Q.4, if sampling were limited to the region of the dashed circle, the derived values might not be appropriate to simulate the overall performance of the site. It is important to consider the scale of the variability (e.g., the correlation length) compared to the scale of the disposal facility. Short correlation lengths in comparison to the scale of the facility are not generally a cause for concern. It can be especially important to ensure that statistical techniques used to develop data have preserved real observed variability.

### *Q.5.2.2 Data Correlations*

Since PAs attempt to simulate many different processes (e.g., infiltration, waste release, transport), it is important to assess the need for correlation of different parameters. For computational reasons, PAs commonly use abstractions. An abstraction is a simplification of a process model that represents the essential elements of the process model. The abstractions or submodels of a dose or performance assessment may not directly simulate global processes but can be commonly influenced by a global process or condition. If correlations should have been used but were not, the net effect is that the likelihood of favorable or unfavorable conditions may not be appropriately represented. Two examples are used below to illustrate the concept.

The first example is for consumption data in a biosphere model. Consumption data are needed when food pathways are considered in an assessment. Usually data are provided for the individual consumption rate of different items (e.g., beef, pork, chicken, eggs, milk), using an average member of the critical group construct. Problems can arise if the analyst does not impose correlation in the data or constraints on consumption. The average member will have a defined range of caloric intake that should be consistent with the assumptions about the individual. If all consumption parameters are sampled independently, at the extremes it can result in nonphysical overall caloric or fluid intake rates, both high and low.

The second example involves corrosion of a carbon steel disposal container and transport through underlying soil. The corrosion rate of the carbon steel could be input into the dose or performance assessment as a simple rate (mils/y)[1] that characterizes the observed values of corrosion rates. Likewise, the transport rate through the soil can be characterized by a retardation factor, which is, itself, a function of a distribution coefficient. Both of these seemingly different processes will be influenced by moisture content and pH. Independent simulation of the processes without correlation in the parameters (analogous to a lumped parameter model) may result in nonphysical results. For instance, the corrosion rate may be higher, and the distribution coefficient may be lower at low pH values. If pH was sampled and both the corrosion rate and distribution coefficient were correlated to it, the response of the system model could be appropriately captured. If state variables are defined and the submodels or abstractions are based on the state variables, the need for complex correlation structures can be reduced or eliminated.

[1] 1 mil = 0.001 inches

### *Q.5.2.3 Use of Generic Data*

Generic data are commonly used in PAs. In most applications, the goal of the dose or performance assessment is to provide estimates of site-specific performance. As discussed in Section Q.4.2.1, the representativeness of generic data must be considered. A generic dataset may contain—in fact, is likely to contain—values that are not representative of site-specific conditions. Some parameters contribute mainly to the magnitude of the output metric (e.g., drinking water ingestion rate), whereas others can contribute to both timing and magnitude (e.g., Kd values). If the generic data are unbiased compared to the site-specific data and only affect the magnitude, the impact is mainly on the perceived variance in the output but not on the mean of the output. However, skewed generic data can lead to unreliable estimates.

Generic data used in site-specific PAs that can influence the magnitude and timing of the output require careful consideration. Figure Q.5 shows a hypothetical distribution coefficient ($K_d$) parameter distribution derived from a generic reference such as “Default Soil Solid/Liquid Partition Coefficients, Kds, for Four Major Soil Types: A Compendium,” (Sheppard and Thibault, 1990). The generic distribution may represent all measurements performed over all sites. It contains within the distribution all observed sources of uncertainty and variability, such as mineralogy, geochemistry (pH, Eh), and other sources of variability (e.g., measurement error, measurement technique). The color-coded symbols represent “measurements” at three different sites. The sites may have different mineralogy (sand, loam, clay) and different geochemical conditions. Inclusion of the portion of the generic distribution that is not representative of the site-specific conditions can result in risk dilution. For instance, the sorption associated with the “x” symbols in the figure is much stronger on average than the sorption associated with the square symbols. If the specific site being analyzed has conditions comparable to the squares, inclusion of the other data “dilutes” the risk in the analysis. For the analyst, the challenge is determining when generic datasets have been used appropriately and when they have not been.

Different approaches can be used to understand the impact of using generic datasets. As discussed in Section Q.6, sensitivity and uncertainty analyses can identify which parameters are risk significant. If a parameter derived from a generic dataset is identified as being risk significant, a number of actions should be considered:

- Collect site-specific information to replace or constrain the generic dataset.
- Introduce conservative bias in the representation of the dataset in the dose or performance assessment.
- Use an alternative metric for decision-making, such as mean of the peaks, rather than the normal probabilistic model output metric, which is peak of the means.

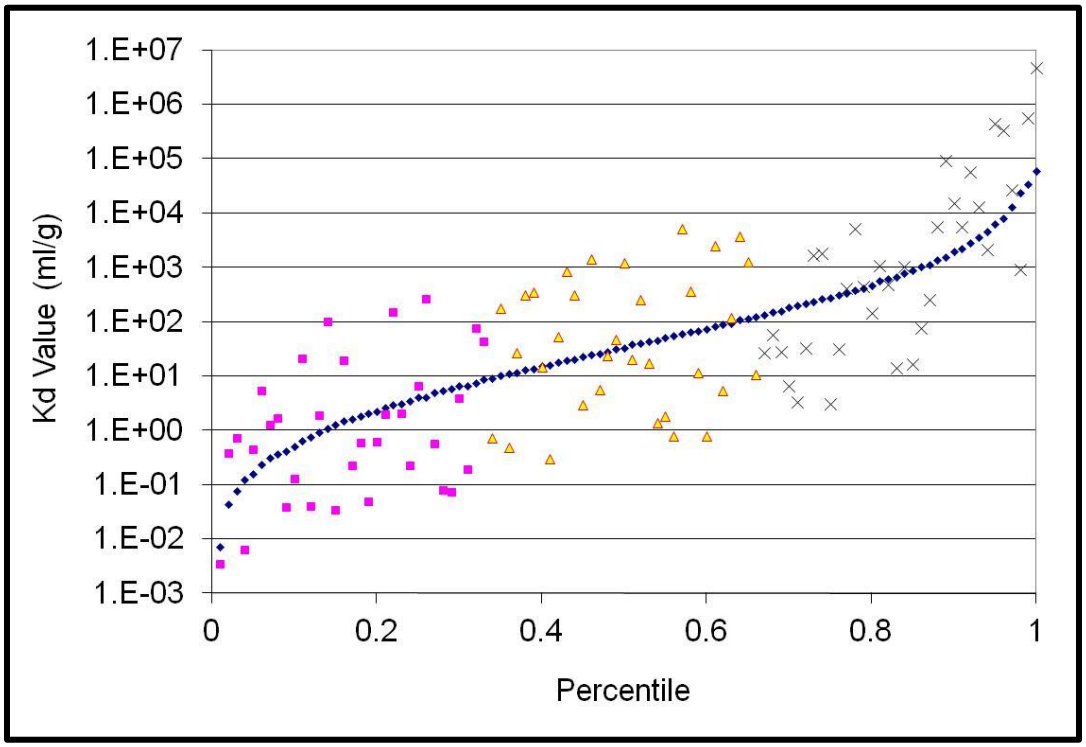


**Figure Q.5 $K_d$ Distribution with Values Plotted from Measurements at Three Different Hypothetical Sites**

Figure Q.6 is from a paper entitled, “The Impact of Uncertainty Type and Representation in Performance Assessment,” 2012 (Esh, Grossman, and Parks, 2012), which used a simplified dose or performance assessment model to examine the impact of sparse information and generic datasets on assessment model output for different analysis techniques. As shown in (a), risk dilution can occur in probabilistic modeling using generic datasets for a site-specific evaluation. In addition, when information is sparse to establish parameter distributions, there is a higher likelihood that the distribution assigned may not be representative and may be in error. The analyst should consider how much information is available to assign parameter distributions and how representative it may be.

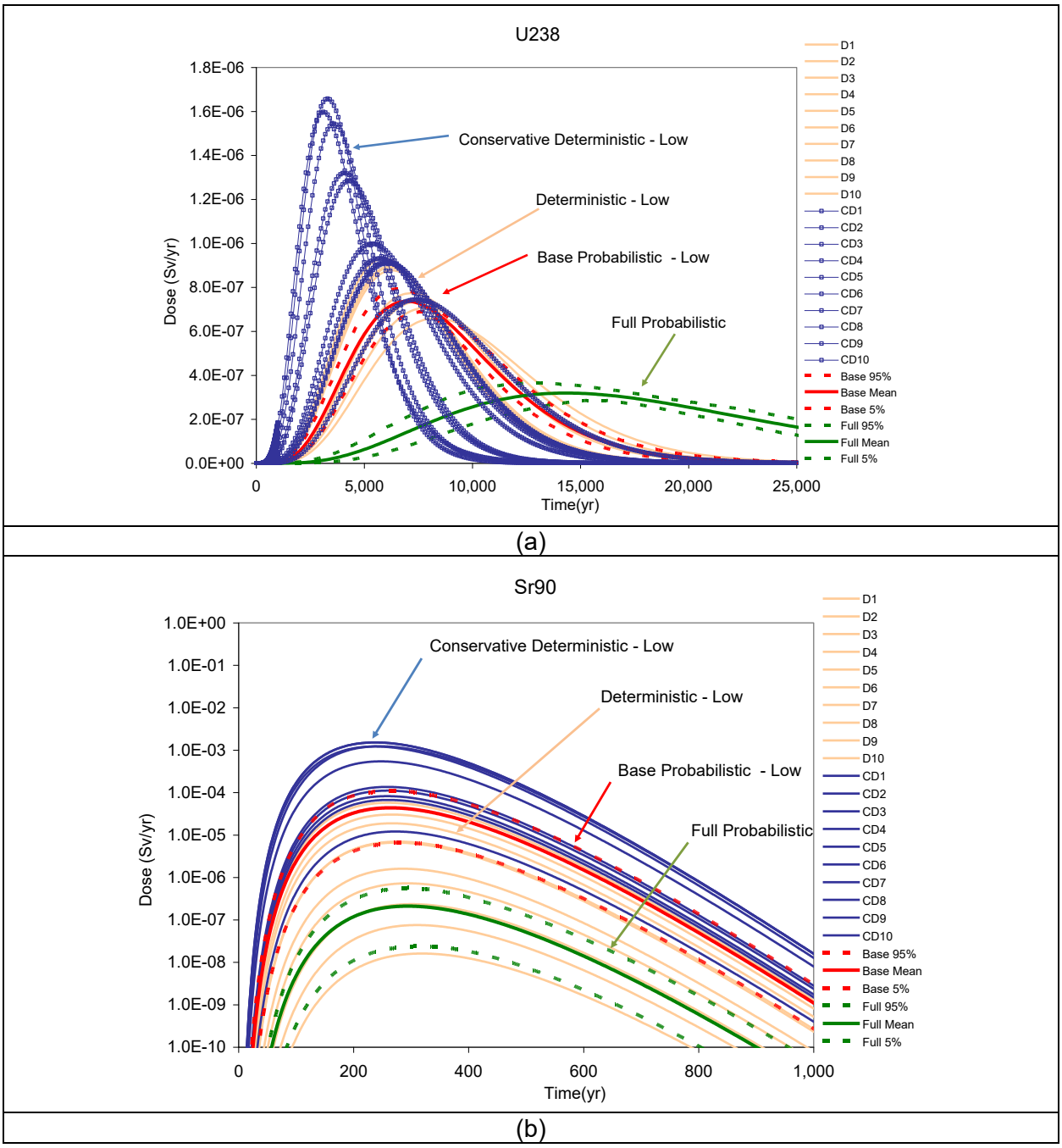


**Figure Q.6** **Influence of Different Amounts of Site-Specific Data on Performance Assessment Model Output** *(a) is for Uranium whereas (b) is for Sr-90; due to Risk Dilution, even a Very High Percentile of the Probabilistic Modeling Results may be outside of the Range Expected for a Site-Specific Analysis*

### *Q.5.2.4 Alternative Conceptual Models*

Model uncertainty is common in system modeling, especially for complex systems. Dose and performance assessment models cannot be validated in the traditional sense, which results in model uncertainty. Even the submodels of a dose or performance assessment, which may be focused on select aspects of the overall modeling effort, usually cannot be validated because of the projections over very long timescales. Model support may vary between submodels such that some models have high confidence whereas others have low confidence. Model uncertainty is directly tied to model support; when model support is limited, it can be expected that model uncertainty will be higher.

Even with a strong effort to develop model support, there may be multiple conceptual models that are consistent with the available supporting information. Table Q.1 provides an example of a dose or performance assessment model broken down into submodels. In this example, there are multiple submodels that could be used to represent different parts of the dose or performance assessment model. There are a total of 24 different combinations of submodels. A challenge arises because the relative importance of an individual submodel may or may not be conditional on the other submodels with which it is combined. For example, waste release submodel 1 (WR1) may be of relative importance[2] 1 when combined with solubility 1 (SOL1) but could be of relative importance 3 when combined with solubility 2 (SOL2).

**Table Q.1 Example of Alternative Conceptual Models in a Performance Assessment**

| Submodel | Alternative 1 | Alternative 2 | Alternative 3 |
|---|---|---|---|
| Infiltration | INF1 | INF2 | |
| Waste Release | WR1 | WR2 | WR3 |
| Solubility | SOL1 | SOL2 | |
| UZ Transport | UZ1 | UZ2 | |
| SZ Transport | SZ1 | | |

Different methods can be used to assess the importance of alternative conceptual models. The analyst should base the safety decision on a conceptual model that is adequately supported or sufficiently conservative to account for the effects of model uncertainty on the decision. Three methods to address model uncertainty are the following:

(1) iterative dose and performance assessment using increased model support on risk significant submodels to reduce model uncertainty

(2) analysis of submodel combinations; use of conservative combinations

(3) probability weighting of submodels based on the degree of belief of each submodel and performing a probabilistic assessment

All of these methods require some amount of quantitative representation of each submodel, although abstractions may be used to reduce the resources required to perform the assessment. The preferred method to deal with model uncertainty is to use an iterative dose or

[2] Relative importance is defined as the ratio of risk using the submodel compared to the relative risk using the submodel that provides the lowest risk.

performance assessment. First, sensitivity and uncertainty analyses combined with barrier analyses are used to identify the most risk-significant submodels (note: risk significance can result in an increase or decrease to the output metrics of concern). Next, model support is developed to constrain the conceptual models and identify which submodels are most accurate. If enough model support can be developed to identify the most accurate submodels, then model uncertainty has been effectively reduced.

The second method for determining model uncertainty is to evaluate each combination of submodels individually and evaluate the results. The most conservative combination likely will not reflect "risk," but it may provide a bounding argument to make a regulatory decision. This method avoids the problem of assigning a degree of belief to the different submodels. This method can also be used by selecting a conservative combination for decision-making purposes, although the conservatism is subjective and should be presented as such.

The third method is similar to the second except a degree of belief is assigned to each submodel. This makes the importance of the submodels more risk informed, if the degrees of belief are based on an unbiased process. The challenge is that, when model support is limited, the subjective degrees of belief may have significant error rates associated with them. It is advisable to include uncertainty in the degrees of belief, however they are obtained. One method is to perform independent expert elicitation.

*Q.5.2.5 Model Integration*

There may be uncertainty associated with how submodels should be integrated, especially when model support is limited. Model integration involves developing the connections between submodels by defining what information is passed and how it should be passed. Ideally, the connections between submodels should be based on observable information (e.g., concentration of radionuclides, flux of radionuclides) but that may not always be possible, especially for a facility under design. The information that needs to be passed is primarily addressed initially by the subject matter experts developing the model and is confirmed by model support activities.

It is important to ensure that the spatial and temporal distribution of the information being passed is at an appropriate resolution such that data reduction has not eliminated important model response. The following example shows the flow of chloride concentrations through three different submodels (soil layer, engineered cover, waste zone) in a waste disposal system. The ultimate goal of the model is to estimate the corrosion rate of a steel liner located below the waste zone. The corrosion rate is a function of the chloride concentration:

$C\ (\text{mils/y}) = a\ (Cl^-)^b$ where a and b are constants

The flux rate and concentration of chloride is represented with ten cells in the soil submodel (see Figure Q.7); the concentrations vary from cell to cell. Those concentrations (fluxes) are translated through the system until they are used to estimate failure times of the steel liner. The estimated failure times may not be accurate if the temporal and spatial resolution of the data has not been preserved during submodel integration. Assuming the chloride concentration at location 1 is five times the value in the neighboring cell (1.1) in the top soil layer, the concentration of chloride assuming only vertical flow will be approximately 1.7 times less when it reaches the liner (region 4).

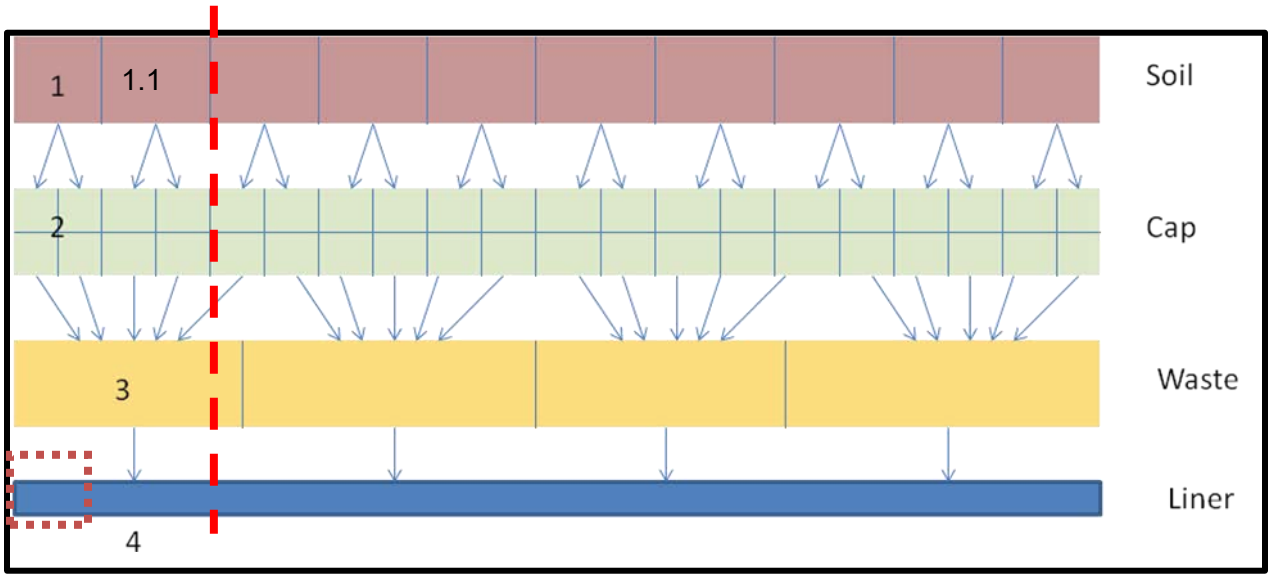


**Figure Q.7 Hypothetical Waste Disposal System Showing Representation of Different Submodels**

For the expression given above, if b = 1.7, then the total corrosion estimated at region 4 would be approximately 68 mils (from the red dashed line to the left), compared to 170 mils in the highlighted region. In one case, failure may not occur whereas, in the other case, releases may occur, albeit over a smaller area. This example highlights the importance of proper spatial and temporal integration of submodels. Comparison with observed values can help ensure that the temporal and spatial resolution was appropriate for the problem.

*Q.5.2.6 Abstractions*

It is expected that abstractions may be needed in dose or performance assessment modeling. An abstraction is a simplification of a process model that represents the essential behavior of the process model. Examples include but are not limited to lookup tables, response surfaces, and parameter distributions. An abstraction preserves the essential behavior in a form that can be implemented in a computationally efficient manner. The use of abstractions can result in reduced complexity that has many benefits, such as easier review and interpretation. The main disadvantage is that complex responses of integrated models may not be preserved. The questions a reviewer may ask with respect to abstractions are found in Section Q.4.1.

The following example is used to illustrate abstractions in the form of parameter distributions and how model performance may be improved by ensuring the abstractions are consistent with the underlying state variable. Consider a system in which the solubility of a radionuclide and the distribution coefficient are represented with a parameter distribution. The underlying model response is equivalent to:

Y = a (solubility) / (Kd)

However, both solubility and the distribution coefficient are driven by the underlying state variable pH in the example. Figure Q.8 provides the model response as a function of Kd for two different representations: (1) the abstractions for solubility and Kd (parameter distributions) are

based on the observed ranges and are sampled independently, also defined as a Level 0 abstraction, and (2) the abstractions for solubility and Kd still use parameter distributions but those distributions are based on an underlying state variable pH (in effect the parameter distributions are correlated), which is known as a Level 1 abstraction.[3] As can be seen from the figure, the overall model response is more constrained for higher order abstraction. It may not always be possible to define higher order abstractions, because the state variables themselves or the responses to the state variables may not be known. However, if higher order abstractions can be defined, they should be used, as they preserve more of the underlying process model response.

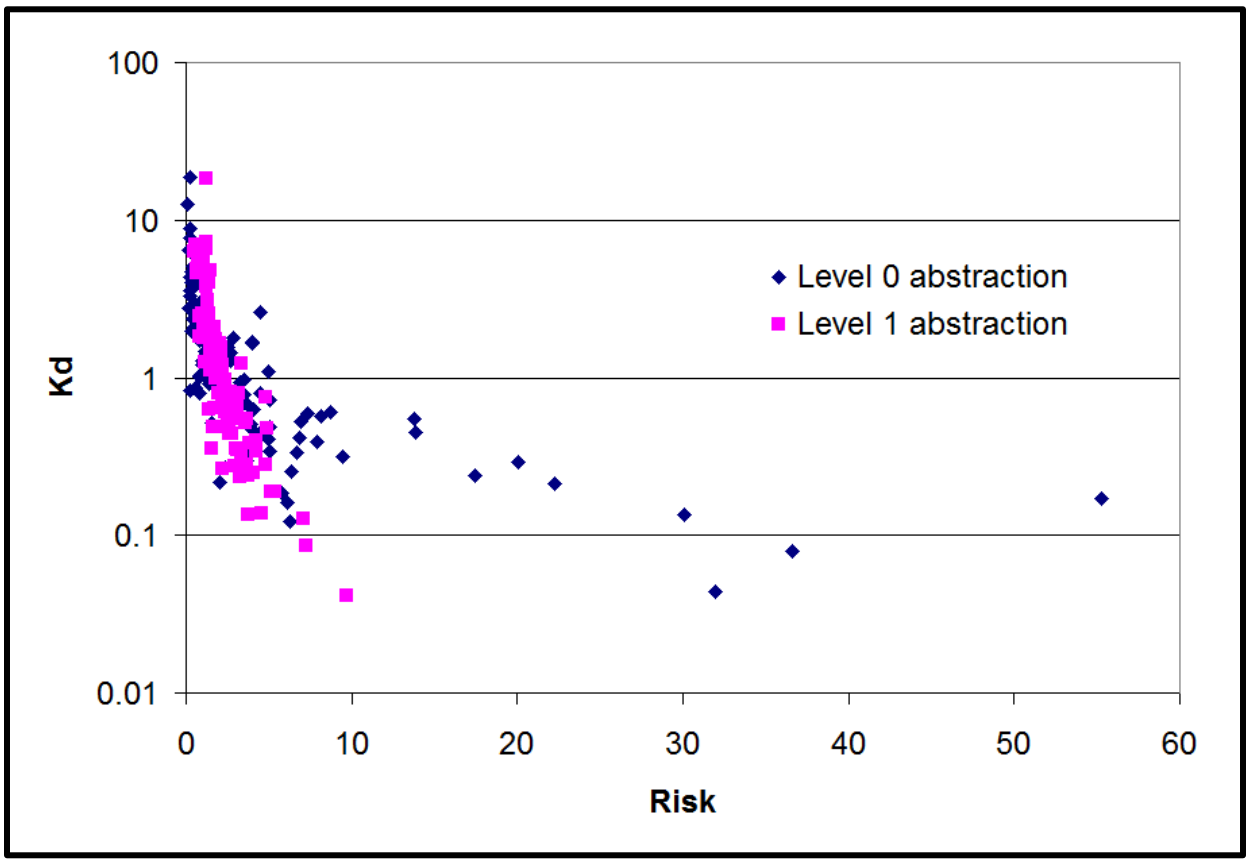


**Figure Q.8 Model Response as a Function of Kd Using a Level 0 or Level 1 Abstraction**

## Q.6 Sensitivity and Uncertainty Analyses

Sensitivity and uncertainty analyses can and should be used to understand dose or performance assessment models; to identify risk-significant parameters, models, and assumptions; and to evaluate the incorporation of uncertainty in the assessment. In the high-level waste program, the NRC has over 20 years of experience performing sensitivity and uncertainty analyses (CNWRA, 2011). The Center for Nuclear Waste Regulatory Analyses (CNWRA) developed a compilation of the different techniques and the lessons learned associated with the use of those techniques. That report should be considered as an essential

[3] In the Level 1 abstraction, the $K_d$ and solubility are both a function of the state variable pH, which is uncertain.

reference to complete or review sensitivity and uncertainty analyses of PAs. The report includes sections on:

Uncertainty Analysis—Provides discussion on topics such as parameters, sampling, maximum entropy, correlations, convergence, output metrics, and risk dilution.

Sensitivity Analysis—Provides discussion on sensitivity analyses applied to parameters, barriers, and submodels. It discusses traditional regression-based methods, as well as nonparametric methods.

Advanced and Special-Case Sensitivity Techniques—Provides a detailed summary of many methods that have been developed over time to support the high-level waste dose or performance assessment. However, a fair number of the techniques have been used in other programs. Techniques include but are not limited to Cumulative Distribution Function, Sensitivity Analyses Methods, Genetic Algorithms with Cascaded Variable Selection, Regionalized Sensitivity Analyses, Fraction Factorial Methods, and many more.

## Q.7 Additional Resources

A number of resources cover the topic of uncertainty applied to PAs. This guide is intended to complement those resources. For additional information, the user may want to consult the following, which are listed in the reference section:

**NUREG-1854 (2007)**—Provides review procedures for data uncertainty, model uncertainty, and model support (among other topics) applied to the review of U.S. Department of Energy waste determinations. Specific review procedures associated with individual topics are provided (e.g., radionuclide transport) as well as generic review procedures for PAs.

**NUREG-1573 (2000)**—Provides recommendations of the NRC's Performance Assessment Working Group on acceptable methodologies for PAs of low-level radioactive waste disposal facilities. Section 3.2.4. contains guidance on the treatment of sensitivity and uncertainty in low-level waste. Section 3.3.2 includes a description of sources of uncertainty, issues, and recommended approaches.

**Mohanty, S. et al (2011)**—Provides a robust compilation of sensitivity and uncertainty analyses techniques that have been used for over two decades in the high-level waste program. It includes many lessons learned and valuable tips that are generally applicable to other systems.

## Q.8 References


Esh, D., C. Grossman, and L. Parks, "The Impact of Uncertainty Type and Representation in Performance Assessment," 11th International Probabilistic Safety Assessment and Management Conference, Helsinki, Finland, 2012.

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C.S. Barr, G.C. Chapman, J.P. Clements, D.W. Esh, R.W. Fedors, M.M. LaFranzo, D.B. Lowman, C.A. McKenney, L.L. Parks, D.W. Schmidt, A.L. Schwartzman, and B.A. Watson

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11. ABSTRACT (200 words or less)

The U.S. Nuclear Regulatory Commission (NRC), Office of Nuclear Material Safety and Safeguards (NMSS), previously consolidated and updated numerous decommissioning guidance documents into a three volume NUREG. This NUREG series is intended for use by the NRC staff, licensees, and others. The three volumes address the following topics:
(1) Decommissioning Process for Materials Licensees
(2) Characterization, Survey, and Determination of Radiological Criteria
(3) Financial Assurance, Recordkeeping, and Timeliness

The staff last updated Volume 2 of the NUREG series, entitled, "Consolidated Decommissioning Guidance: Characterization, Survey, and Determination of Radiological Criteria," in September 2006. This volume provides guidance on compliance with the radiological criteria for license termination (License Termination Rule (LTR)) in Title 10 of the Code of Federal Regulations (10 CFR) Part 20, "Standards for Protection against Radiation," Subpart E, "Radiological Criteria for License Termination." This guidance takes a risk informed, performance based approach to the demonstration of compliance. This guidance will help identify the information (subject matter and level of detail) needed to terminate a license and considers the specific circumstances of the wide range of NRC licensees. Licensees should use this guidance in preparing decommissioning plans (DPs), license termination plans, final status surveys, and other technical decommissioning reports for NRC submittal. The NRC staff will use the guidance in reviewing these documents and related license amendment requests. Volume 2 applies to all licensees subject to the LTR (i.e., fuel cycle, fuel storage, materials, and reactor licensees).

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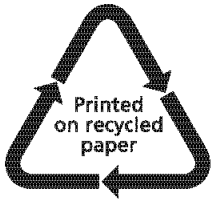


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