Lessons Learned and Questions and Answers to Clarify Decommissioning and License Termination Guidance and Plans

Note: This document was extracted from NUREG-1757, Volume 2, Revision 1, Appendix O. The information has been incorporated into the text of NUREG-1757, Volume 2, Revision 2, as applicable, and the information in this document is no longer found as a separate appendix. In some cases, additional guidance on technical topics discussed in this document was developed in updating NUREG-1757, Volume 2. In those cases, the latest guidance found in NUREG-1757, Volume 2, would apply.

This document archives previous lessons learned and questions and answers found in NUREG-1757, Volume 2, Revision 1, Appendix O, which attempted to clarify guidance associated with the decommissioning process and the License Termination Rule (LTR). Specifically, it provides (a) questions and answers; (b) lessons learned by NRC) staff during the review of submitted decommissioning plans (DPs) and license termination plans (LTPs); and (c) lessons learned from decommissioning final status survey inspections and follow-up confirmatory surveys. Some of the information provided in this document was originally developed for reactor licensees but is also applicable to other types of facilities.

1.0 Nuclear Energy Institute Questions and Answers to Clarify License Termination Guidance

1.1 Introduction

During a June 1, 2001, public workshop on Nuclear Regulatory Commisison (NRC)'s Decommissioning Guidance Consolidation Project (i.e., this NUREG report series), the Nuclear Energy Institute (NEI) and NRC staff identified an approach to clarify existing guidance associated with the LTR (Title10 of the *Code of Federal Regulations* (CFR) Part 20, Subpart E), in concert with the decommissioning guidance consolidation project. Under this approach, NEI's License Termination Task Force (Task Force) generated questions (Qs) associated with decommissioning issues that are common to the industry. The Task Force also proposed answers (As) to the questions and submitted the Questions and Answers (Q&As) to NRC staff for review. NRC staff reviewed the Q&As and the supporting technical bases and provided comments to NEI on September 28, 2001. An open meeting was held between NRC, NEI, and industry representatives on December 4, 2001, to discuss each Q&A and related technical issues to ensure that the questions were properly asked and answered and were supported by a defensible technical basis. NRC staff and NEI further developed the Q&As so that they adequately reflect NRC regulations and guidance and include a sound technical basis. Nothing in this set of Q&As modifies or negates the guidance presented in NUREG–1700, "Standard Review Plan for Evaluating Nuclear Power Reactor License Terminations Plans"; Regulatory Guide 1.179 , "Standard Format and Content of License Termination Plans for Nuclear Power Reactors"; and the Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM, NUREG–1575). It should be noted that when using the guidance provided in the responses to the questions in the Q&A section of NUREG-1757, Volume 2, in preparing a DP or LTP, the licensee remains responsible for compliance with the LTR, its implementation, and providing the staff with the information necessary to prepare the Safety Evaluation Report and Environmental Assessment.

Seven Q&As found acceptable by NRC staff are provided below.

Question 1: Development of Radionuclide Profiles for Reactor Facilities

In support of the MARSSIM process, radionuclide distribution profiles are necessary to ensure that survey and analysis techniques are appropriate and that dose assessments properly consider all the radionuclides that may be present. During the process of developing initial radionuclide profiles for characterizing commercial light-water reactor sites and facilities, which radionuclides are considered and what resources and methodologies are appropriate?

Answer to Question 1

A unique radionuclide profile must be developed for each of the major types of materials expected to remain onsite after remediation. A commercial light-water power reactor facility will likely require profiles for contaminated soil or sediments, surface contaminated materials, and activated materials. The licensee must consider that activation products in steels and concretes vary with the constituents and operational history. Concrete will also differ between facilities because of different trace elements. While one generic list cannot be developed that would be applicable to all power reactor licensees and types of contaminated materials, once radioactive decay has been considered to the time when final status surveys (FSSes) will be conducted, a set of radionuclides may be developed for surface contamination and for activated materials. The profiles listed below in Table 1 are not meant to be all-inclusive, and other radionuclides should be added, as necessary, based on site-specific considerations.

Table 1 Example Radionuclide Profile

Contamination Suite		Activation Suite
H-3	Sb-125	H-3
C-14	Cs-134	C-14
Mn-54	Cs-137	Fe-55
Fe-55	Eu-152	Ni-63
Co-57	Eu-154	Co-60
Co-60	Ce-144	Cs-134
Ni-59	Pu-238	Cs-137
Ni-63	Pu-239/240	Eu-152
Sr-90	Pu-241	Eu-154
Nb-94	Am-241	Eu-155
Tc-99	Cm-243/244	Mn-54, Ni-59, Zn-65

The licensee should confirm, by using characterization surveys and historical assessments, that the radionuclide lists developed are applicable to the facility and appropriate for each medium. Technical considerations and limitations are discussed in: NUREG/CR–3474, “Long-Lived Activation Products in Reactor Materials”; NUREG–0130, “Technology, Safety and Cost of Decommissioning”; and NUREG/CR–4289, “Residual Radionuclide Contamination Within and Around Commercial Nuclear Power Plants.” Characterization surveys conducted according to NUREG–1575, “Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM),” provide information on the important radionuclides that must be considered. The licensee may also use (a) radionuclide distributions developed for waste classification to demonstrate compliance with requirements of 10 CFR Part 61, and (b) analyses such as ORIGEN computer code runs, to help determine which radionuclides to consider. It is important to recognize the limitations of such methods as they apply to the MARSSIM process. The licensee should also consider historical fuel performance, operational history, and time since shutdown. It is incumbent on the licensee to ensure that the list of radionuclides for each material type is developed according to NRC guidance (such as that in MARSSIM) and using good laboratory practices.

Question 2: Radionuclide Deselection

When developing derived concentration guideline levels (DCGLs) for the final status survey (FSS), which radionuclides can be deselected from further consideration?

Answer to Question 2

Guidance in Section 3.3 of NUREG-1757, Volume 2, states, " [o]nce 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." Therefore, during characterization of a facility, if a profile contains radionuclides that collectively contribute less than 10 percent of the dose criterion, those nuclides may be deselected from the list. Since DCGLs are developed to equate to the radiological criteria for license termination (0.25 mSv/y (25 mrem/y)) TEDE to the average member of the critical group and ALARA, for unrestricted release in 10 CFR 20.1402, those radionuclides that collectively contribute less than 0.025 mSv/y (2.5 mrem/y) may be considered insignificant, provided all appropriate exposure scenarios and pathways are considered. It is incumbent on the licensee to have adequate characterization data to support and document the determination that some radionuclides may be deselected from further detailed consideration when planning the FSS. Radionuclides that are undetected may also be considered insignificant, as long as the minimum detectable concentrations (MDCs) are sufficient to conclude that the dose contribution is less than 10 percent of the dose criterion (i.e., with the assumption that the radionuclides are present at the MDC concentrations). In addition, licensees should note that they are required to comply with the applicable dose criteria in 10 CFR Part 20, Subpart E, and thus the dose contribution from the insignificant radionuclides must be accounted for in demonstrating compliance with the dose criteria.

Question 3: Embedded and Buried Piping Characterization

What are acceptable methods to characterize embedded piping and buried piping?

Answer to Question 3

Several methods have been used to characterize the residual activity within embedded pipe, and these methods can be used for buried piping, as well. By definition, "embedded piping" is piping (e.g., part of a plant system) that is found in buildings and encased in concrete floors and walls, while "buried piping" is piping (e.g., culvert) that is buried in soils. To be found acceptable, the methods should each address the following nine issues:

- radionuclides of interest and chosen surrogate,
- levels and distribution of contamination,
- internal surface condition of the piping,
- internal residues and sediments and their radiation attenuation properties,
- removable and fixed surface contamination,
- instrument sensitivity and related scan and fixed MDCs,
- piping geometry and presence of internally inaccessible areas/sections,

- instrument calibration, and
- data quality objectives (DQOs).

An industry study (Cline, J. E., "Embedded Pipe Dose Calculation Method," Electric Power Research Institute Report No. 1000951, November, 2000) evaluated several techniques for measuring the radiological contamination on the inside of embedded pipe. Measurement techniques included pipe crawlers, gamma-ray scanners, dose rate measurements with dose-to-curie computations, scraping samples with radiochemical analyses, and smear samples with radiochemical analyses. A brief description of these methods is provided below.

- The pipe crawler uses a beta sensitive detection system that is inserted into the pipe with a cable. Spacers keep the detectors at a fixed distance from the pipe wall. Measurements can be made at various points or as a continuous scan within the pipe to provide a profile of the extent and distribution of the contamination. Scaling factors based on a laboratory radiochemistry analysis of the deposited material can be applied to the measurements to provide radionuclide quantities in the pipe.
- The gamma-ray scanner uses a calibrated, collimated high-purity germanium or sodium iodide spectrometer to make external measurements on the pipe. This gamma-ray scanning yields an average concentration over the length of the pipe within the field of view of the detector. The sensitivity of this method may be limited by the thickness of the piping itself and concrete between the pipe and the detector. Some radionuclide identification is possible and scaling factors can be applied as discussed above for the pipe crawler.
- The dose rate measurements are also made on the external surface of the walls or floors containing the embedded pipe using a sensitive gamma detector capable of reading in the roentgen per hour range. The dose rate readings may be used directly in determining compliance with the dose criteria or used to make dose-to-curie conversions based on other measurements providing radionuclide identification.
- Radionuclide identification for the contamination in the pipe may be accomplished by smear or scraping samples and radiochemical analysis. The industry report compared radionuclide ratios determined by smears and by scrapings with those found by etching the surface of the pipe. The report concluded that either of these techniques yields radionuclide mixes that are representative of the average total deposits. Each approach is useful in specific applications and multiple methods might be used in complex facilities like power plants. Each method also has limitations and uncertainties that must be addressed.

Other useful information on embedded pipe characterization may be found in sources such as the U.S. Department of Energy's (DOE) Innovative Technology Reports and case studies published in open literature.

Regardless of the source of the information, it is incumbent on the licensee to develop and document a comprehensive approach for characterizing embedded pipe and buried piping that accounts for limitations and uncertainties, taking into account MARSSIM guidance in developing the related DQOs. It should also specifically address each of the critical issues in the bulleted list above.

Question 4: Development of Site-Specific Distribution Coefficient Values for Soil or Concrete

What is an acceptable approach for the development of input distribution coefficient (K_d) values for soil or concrete when using site-specific dose modeling codes?

Answer to Question 4

K_d values for input into site-specific dose modeling codes may be determined by the following:

Use sensitivity analyses, which include an appropriate range of K_d values, to identify the importance of the K_d to the dose assessment and how the change in K_d impacts the dose (i.e., how dose changes as K_d increases or decreases). The range of K_d values that bound the sensitivity analysis may be obtained from (a) the literature, (b) the default distribution in Dand D, or (c) the default distribution in the probabilistic code of RESRAD (please refer to the “Basis” section that follows).

Using the results of the sensitivity analysis, choose a conservative K_d value, depending on how it affects the dose (e.g., if higher K_d values result in the larger dose, an input K_d value should be selected from the upper quartile of the distribution; if lower K_d values result in the larger dose, an input K_d value should be selected from the lower quartile of the distribution). For those isotopes where the K_d does not have a significant impact on the dose assessment (i.e., K_d is not a sensitive parameter), the median value within the range is an acceptable input parameter.

If the licensee feels that the K_d value is overly conservative, the licensee is encouraged to perform a site-specific K_d determination so that the dose assessment reflects true site conditions.

Basis

The licensee is encouraged to use sensitivity analyses to identify the importance of the K_d parameter on the resulting dose either (a) to demonstrate that a specific value used in the analysis is conservative or (b) to identify whether site-specific data should be obtained (if the licensee feels K_d is overly conservative). The sensitivity analysis should encompass an appropriate range of K_d values. As noted above, the input range for the sensitivity analysis may be obtained from literature, DandD default distribution, or RESRAD probabilistic default distribution.

Literature

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, are universally appropriate in all cases for determining appropriate K_d values. Although K_d values are intended to represent adsorption, in most cases they are an aggregate 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.

DandD

The use of the default K_d values from the most recent version of the DandD code outside of the scope of DandD may not be justified since the single set of default parameters derived for DandD were developed assuming a specific set of exposure pathways and a specific source term. Any single parameter value taken from the default set of parameters outside of the context of the given exposure scenario, source term, and other parameters will have no meaning in terms of the original prescribed probability; therefore there is no basis to conclude that any default K_d value will give a conservative result. However, the *distribution* of K_d values used in DandD (which can be found in NUREG/CR–5512, Volume 3, "Residual Radioactive Contamination from Decommissioning—Parameter Analysis," Table 6.86) can be used as the range of K_d values for the sensitivity analysis.

RESRAD

RESRAD default parameter values (including K_d values) should not be used. The default values were included in the code primarily as place holders that enable the code to be run; it was assumed that site-specific values would be developed. However, it is appropriate to use the default parameter distribution developed for the RESRAD family of codes as the range for use in the sensitivity analysis.

After performing sensitivity analysis with the appropriate K_d ranges, the K_d value at the upper or lower quartile of the distribution, which ever results in the highest derived dose, can be considered an acceptable value to use in the dose code; no further justification is required[1]. For those K_d values that are overly conservative, a site-specific K_d value may be determined by the direct measurement of site samples. Appropriate techniques for K_d determination include American Society for Testing and Materials (ASTM) and U.S. Environmental Protection Agency (EPA) methods 9–83, "Distribution Ratios by the Short-Term Batch Method"; ASTM D 4646–87, "24-h Batch-Type Measurement of Contaminant Sorption by Soils and Sediments"; and "Understanding Variation in Partition Coefficient, K_d Values," Volumes I and II, EPA 402–R–99–004A, available at https://www.epa.gov/radiation/understanding-variation-partition-coefficient-kd-values.

Question 5: Demonstrating Appropriate Selection of Survey Instrumentation by Illustrative Example

Is it acceptable to define (a) the DQO process and (b) the acceptance criteria for demonstrating that the radiation survey instrumentation selected for use in the FSS are sufficiently sensitive for a given DCGL and expected survey conditions using illustrative examples?

Answer to Question 5

Yes, it is acceptable to define the DQO process and acceptance criteria using examples that demonstrate the appropriate selection of radiation survey instrumentation for the expected types of FSS surface conditions and radionuclides forming the basis of the DCGL.

For example, the selection of instrumentation may be grouped by category of surfaces with similar features and expected instrument responses over these surfaces. For each of the

[1] Updated guidance on this topic is found in Appendix I of NUREG-1757, Volume 2, Revision 2.

defined categories of survey instrumentation and methods presented in the LTP (e.g., soil scanning, surface scanning and surface fixed measurements), the licensee should provide the derivation of scan and fixed MDCs. The derivation of the MDCs must take into account instrument efficiencies (surface and detector), scan rates and distances over surfaces, surveyor efficiency, and minimum detectable count rate using the guidance in MARSSIM and NUREG-1507, "Minimum Detectable Concentrations with Typical Radiation Survey Instruments for Various Contaminants and Field Conditions."

Instruments, other than those provided as examples in the LTP, may be used for the FSS as long as the process approved in the LTP is used to show that the substitute instrument has equal or better performance. If a licensee were to use new technologies (e.g., *in situ* gamma spectroscopy) or different instrumentation than those that were considered at the time of the LTP submittal, the new technology or instrumentation must be shown to perform with sensitivities that allow detection of residual radioactivity at an appropriate fraction of the DCGL and corresponding investigation levels. In addition, the new technology or instrumentation must be at least as efficient as examples of survey instrumentation provided in the LTP. A licensee should also demonstrate and document that conducting the FSS by this new method will meet all related DQOs for demonstrating that survey units meet the site-established DCGLs.

Question 6: Characterization of Items to be Removed Prior to License Termination

Is the collection of additional characterization data, beyond that available from periodic radiation protection surveys, required in the LTP for structures, components, and soils that will be removed from the facility prior to license termination?

Answer to Question 6

No. In general, radiological data obtained during characterization surveys are used to determine the radiological status of the site, including facilities, buildings, surface and subsurface soils, and surface and groundwater. In turn, this information is used to support the planning and design of the FSS. In addition to providing the basis for the design of FSS, characterization surveys are used to support the following:

- Identification of remaining site dismantlement activities,
- Development of new (or revisions to existing) remediation plans and procedures,
- Revisions to decommissioning costs and trust fund,
- Identification of environmental aspects not previously considered,
- Revisions to the Environmental Report.

Since the license termination process is only concerned with the status of facilities after the completion of all remediation activities, radioactivity associated with structures, components, and soils that will be removed from the facility and appropriately disposed of elsewhere, is not an issue as it cannot contribute to the site's public dose controlled under 10 CFR 20.1402 – "Radiological Criteria for Unrestricted Use." Therefore, additional characterization data need not be collected.

Question 7: Characterization for Initial Classification of Class 1 Areas

Is characterization data required to support initial classification of Class 1 areas?

Answer to Question 7

Areas classified as Class 1 do not require characterization data to support that classification.

Note that characterization data are needed to support decommissioning activities for all areas including:

- Determination of radionuclide distribution profiles and identification of surrogate radionuclides,
- Dose modeling and development of DCGLs,
- FSS design and instrument selection,
- Structuring the DQOs,
- Assessment of spatial variability of radioactive contaminants on building surfaces and in surface and subsurface soils,
- Assessment of whether groundwater is impacted, using the results of the surface and subsurface soil characterization surveys,
- Initially defining and changing the boundaries of Class 1 survey units with bordering and adjacent survey units,
- Reclassification of survey units (using guidance in MARSSIM and Section A.2 of Appendix A of NUREG-1757, Volume 2).

2.0 RIS 2002-02, "Lessons Learned Related to Recently Submitted Decommissioning Plans and License Termination Plans"

2.1 Introduction

Since the implementation of the LTR, NRC staff has reviewed several DPs and LTPs. As a result of these reviews, the NRC staff learned several lessons, the details of which are discussed in the Regulatory Issue Summary (RIS 2002–02), "Lessons Learned Related to Recently Submitted Decommissioning Plans and License Termination Plans." The information in this section is taken directly from the RIS and is provided to help materials and reactor licensees develop more complete DPs and LTPs, as appropriate. There has been some minor changes in this section relative to the RIS, mainly to provide the appropriate reference to updated sections of NUREG-1757, Volume 2.

2.2 Lessons Learned

The issues concerning lessons learned include the following ten lessons.

Lesson 1: Communications

Early and frequent consultations between NRC staff and licensees are encouraged during the planning and scoping phase supporting the preparation of the DPs or LTPs. In this context, a licensee may schedule a meeting with the NRC license reviewer assigned to the site to discuss the planning and content of the DP or LTP. The discussions would address (among other topics) past and current licensed operations; types and quantities of radioactive materials used or stored; activities (current or past) that may have an impact on decommissioning operations;

decommissioning goals (restricted versus unrestricted license termination); basis for cleanup criteria and development of site-specific DCGLs or commitment to use NRC default DCGLs; potential impact on public health and safety or the environment; funding plan and financial assurance; and the minimum information required to be contained in the DP or LTP. Regarding the aforementioned topics, licensees are encouraged to review the three volumes of NUREG-1757. The principal purpose of NUREG-1757 is to provide guidance on review of DPs. However, the guidance also supplements NUREG-1700, "Standard Review Plan for Evaluating Nuclear Power Reactor License Terminations Plans," in such areas as site characterization, dose modeling, final radiation survey, and institutional controls. NUREG-1757 provides a structure, using various sections, with which to provide information for staff review. Each section addresses very specific elements of the decommissioning process and related data and information needs. Given that NUREG-1757 presents the information in a generic context, it is the responsibility of the licensee to go over each section and determine which technical elements or regulatory requirements apply to the facility. Appendix D of Volume 1 of NUREG-1757 provides a checklist ("DP Evaluation Checklist") to facilitate this process. Given that the checklist is a brief summary of the material presented in each section, it is recommended that each section be reviewed to gain a full understanding of the requirements as the checklist is being prepared.

Before meeting with the NRC staff, a licensee is encouraged to prepare a checklist that identifies technical elements that are applicable (based on a preliminary review); areas that require clarifications from the NRC staff before decisions can be made as to their applicability to the site or facility; and the scope and level of technical details addressing technical elements and regulatory requirements. In addition, the licensee may wish to make a brief presentation describing the past and current use of the facility and the most current radiological status. During the meeting, the NRC staff and licensee representative would go over each item of the checklist and address specific questions. NRC staff would present an overview of its review process, including discussions of the time line and major milestones. The end product of the meeting is a marked-up checklist that defines the technical elements and regulatory requirements to be covered in the DP or LTP submittal. The NRC staff expects that this process will result in a better understanding of the type of information to be included in either document and to familiarize the licensee with the process that NRC staff will use to evaluate the information contained in the DP or LTP. This approach is expected to minimize the need for requests for additional information, reduce the number of iterations and submittals, and expedite NRC staff's technical review.

Lesson 2: Groundwater

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 ground water at the facility, as well as the extent of this contamination." The actual number, location, and design of monitoring wells depend on the size of the contaminated area, the type 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 ground water 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 large number of suitably designed and installed monitoring wells may be necessary. Power reactors normally have ground water 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 types and the movement of radioactive material contamination in groundwater at the facility, as well as the extent of this contamination. Therefore, a licensee may need to gather additional data to address this lack of understanding.

Lesson 3: Data Quality Objectives

In developing the FSS design, the licensee should 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 cleanup criteria have been met using statistical tests. In brief, the major elements of the DQO process include the following:

- a clear statement of the problem (i.e., a full understanding of the radiological status of the facility and extent and magnitude of the contamination);
- the identification of all related decision statements and alternative actions, including selection of the most appropriate scenario for the site and objectives (i.e., How will compliance be demonstrated?);
- the identification of the information needed to support the decision-making process, such as radionuclide distributions and concentrations, methods used to obtain the data, etc.;
- the definition of the site's physical, temporal, and spatial boundaries for all environmental media and structures, including reference areas, that will be covered by the decision process and modeling;
- the development of a decision rule in defining action levels (e.g., DCGL-wide area ($DCGL_W$)), DCGL–elevated measurement comparison ($DCGL_{EMC}$), MDCs, grid size and layout; statistical tests, and hypothesis;
- specification of limits for Type I and II decision errors in support of the null hypothesis and impacts on sample size and the use of prospective and retrospective power curves; and;
- optimization of the data collection process and updating the design of the survey plan, while meeting all DQOs.

In purpose and scope, the DQO process can include a flexible approach in planning and conducting surveys and for assessing whether survey results support the conclusion that the release criteria have been met. The DQO process can be an iterative process that continually reviews and integrates new information, as needed, into the design of the FSSP and decision-making process. Finally, the selection and optimization of DQOs will facilitate the later evaluation of survey results and decision-making processes during the data quality assessment phase. The NRC staff has observed that licensees have had difficulties in developing DQOs and have not taken full advantage of the DQO process, especially the optimization step. Experience has shown that the process is often rigidly structured by relying too much on characterization data and not being readily open to the possibility of incorporating new information as it becomes available. This approach makes the implementation of any changes difficult and is an inefficient use of resources, since it imposes time delays when determining how to implement any changes.

Lesson 4: Inspections

In-process inspections are more efficient than one-time confirmatory surveys. In one case, the confirmatory survey was conducted after the licensee had completed most of the FSS and many of the staff supporting the final survey were no longer available to address questions and issues that were discovered while conducting the confirmatory survey. Simply put, the confirmatory survey was conducted too late in the process.

The in-process approach has allowed the licensee and NRC to take side-by-side measurements, compare instrument readings and sensitivity, and address survey issues early in the process rather than at the end of the process. The in-process approach has resulted in significant savings in cost, assured a more accurate survey, and helped the licensee in maintaining the release schedule.

Lesson 5: Flexibility

Continued communications between NRC staff and the licensee during the NRC staff's review is to help ensure that the licensee is able to take full advantage of the inherent flexibility in MARSSIM and the three volumes of NUREG-1757. In reviewing DPs and LTPs, the NRC staff has observed that licensees are often boxing their approaches into rigid structures and formats, thereby locking out any operational flexibility in implementing MARSSIM and negating cost savings. This approach may reflect, in part, the interpretation of NRC guidance as regulatory requirements. However, it is possible to meet NRC requirements, while instilling operational flexibility into the overall decommissioning process. For example, large waste volumes alone do not necessarily make a remediation project a complex one, assuming that adequate resources are available to accommodate the higher disposal cost. Aspects of a decommissioning project that make it complex includes such considerations as groundwater contamination; the presence of hard-to-detect and transuranic radionuclides (TRU); heterogeneous distributions of contaminants; the presence of mixed waste; onsite disposal using engineered features; and reliance on institutional controls to maintain doses within NRC limits under restricted-release scenarios, among others. Even under such conditions, there still is an opportunity to simplify the process, maximize operational flexibility, and benefit from economies of scale.

Another example involves how final surveys are structured and designed around survey units, in recognition that some sites may have literally hundreds of survey units, with licensees perceiving that NRC staff needs to approve the FSS design of each one. The NRC staff expects that licensees should group survey units into a manageable number of categories, taking into account the types of buildings, rooms, areas, built-in equipment, and other specific features. This approach is expected to provide the means to identify and address survey unit features and design requirements that are specific for each category, while treating all other common aspects of the survey design in a generic and systematic manner. The NRC staff suggests that the descriptions identify and address, as is applicable, specific survey design requirements, DQOs, sampling methodology, applicable plans and procedures, quality assurance requirements, and data analysis and interpretation for each category. This approach will relieve the NRC staff of having to review and approve each survey design package, before its implementation, and will expedite the final phases of the remediation work, while leaving the development and implementation of each final survey design package subject to periodic regional inspection. Finally, in structuring the final status survey report, licensees should identify and summarize the specific characteristics of each survey unit and discuss their relevance in the analysis of all survey results and interpretation supporting the conclusion that each survey unit meets the cleanup criteria.

Lesson 6: Modeling Issues

The derivation of DCGLs should include the assumptions and justification for parameters used and justification for how these DCGLs will be applied to various survey units onsite. DCGLs will be captured by license condition as part of the LTP approval process, and will require NRC staff approval for changes to the approved DCGLs.

- Area Factors
  Area factors are needed in the FSS to determine the required scan MDCs and to develop $DCGL_{EMC}$ values that are needed to identify small areas that may need further investigation. However, area factors are typically not provided for residual radioactivity on building surfaces. The primary reason is that such factors cannot be calculated using the DandD computer code. Therefore, when screening DCGL values derived from DandD are used an alternative approach must be used to calculate area factors for residual radioactivity on building surfaces.

  One approach that has been successfully used is to develop building surfaces area factors using the RESRAD-BUILD computer code and adjusting these derived area factors to account for the fact that RESRAD-BUILD typically gives less conservative dose estimates. With this approach, the screening DCGL values are converted into the appropriate concentration unit for RESRAD-BUILD (i.e., from "disintegrations per minute per 100 square centimeters" (dpm/100 cm^2) to "pico-curie per square meter"(pCi/m)). Area factors calculated by RESRAD-BUILD can then be adjusted using the ratio of the dose from RESRAD-BUILD to 25 milli-roentgen equivalent man per year (i.e., the equivalent dose from DandD).

- Volumetric Contamination
  Licensees often have volumetric contamination (e.g., contamination below the surface) in the containment structure from activation products. Because the contamination occurs within a building structure, some licensees have assumed that it is appropriate to use DCGL values developed for building surface contamination for these areas without additional justification regarding the appropriateness of their use. However, DCGL values developed for building surface contamination may not be appropriate for areas with volumetric contamination because the potential future exposure routes may be different, especially if the structure is later torn down.

  It is advisable for licensees to develop site-specific DCGL values for volumetric contamination which consider the potential routes of exposure for residual radioactivity in the material for when the structure is eventually torn down. As an alternative, licensees can demonstrate that the DCGL values developed for surface contamination will bound the possible effects from exposures for other configurations of the building structure.

- Model Results
  Licensees using RESRAD, DandD, or other computer codes to generate DCGL values or perform dose analyses often do not include the printout from these codes as part of the decommissioning submittal. This information is typically omitted because the output results tend to be voluminous. However, without this information it is difficult for NRC staff to undertake confirmatory analyses (if needed) or to complete the review of the licensee's analyses.

  It is advisable for licensees to provide output results from any analyses used to develop DCGL values or used to perform dose analyses. If the output results do not provide an echo of the inputs used in the analyses, it may be necessary to also provide copies of the input files.

- Nondispersion Versus Mass Balance Models
  When using the RESRAD computer code to develop DCGL values or to perform dose analyses, licensees often use a nondispersion model for evaluating the groundwater pathways. This model is commonly used because it is the default in RESRAD and therefore will be used unless specifically changed. However, the nondispersion model makes certain assumptions about the location of the future hypothetical well and will generally give lower estimated doses than the mass balance model (if the ground water is an important pathway).

  It is advisable for licensees to either use the more conservative mass balance models or provide justifications for using nondispersion models. Specific guidance on justification for using the nondispersion model can be found in Appendix I of NUREG-1757, Volume 2.

- Parameters
  Licensees often use a combination of default and site-related parameter values in their analyses to develop DCGL values or in dose analyses. In many cases, little or no justification is provided for the specific parameter values used in the analysis. This can lead to uncertainties in assessing the appropriateness of the DCGL values or calculated dose when demonstrating compliance with the standard.

  Given the large number of parameters that may have to be justified in an analysis to develop DCGL values or a dose analysis, Appendix I, Section I.6, of NUREG-1757, Volume 2 discusses an approach for focusing on those parameters most important to the results. This approach entails classifying parameters as either behavioral, metabolic, or physical, as defined in NUREG/CR-5512, Volume 3. Licensees may use default values for behavioral and metabolic (primarily those prescribed for DandD) as long as the values are consistent with the generic definition of the average member of the critical group, and the screening scenarios are used. Site-specific physical parameter values should be used and justified. The level of justification needed is dependent on the significance of the parameter to the results. The relative significance of parameters to the results can be determined through a sensitivity analysis. In the sensitivity analysis, the default statistical distributions provided in RESRAD-ONSITE and RESRAD-BUILD should be used, supplemented with what is known about the site (note: default distributions should not be used as a substitute for known information). Known parameter values should be treated as a constant in the sensitivity analysis. The relative significance of the various parameters can be determined based on the ranks listed in the regression and correlation results in the uncertainty report. The default surface contamination values for alpha-emitting radionuclides are rather low, and in some cases below the detection limit. This results from a conservative resuspension factor (RF) used in the DandD code. Therefore, the licensee may wish to consider using a more realistic RF value for site-specific analyses.

Lesson 7: Decommissioning Cost Estimate

There needs to be a clear relationship between the planned decommissioning activities and the associated cost estimate. At the license termination stage, the NRC staff must make decisions on the proposed actions described in the LTP. The NRC staff typically considers (a) the licensee's plan for assuring sufficient funds will be available for final site release; (b) radiation-release criteria for license termination; and (c) the adequacy of the final survey required to verify that the site release criteria have been met. 10 CFR 50.82(a)(9)(ii)(F) requires the licensee to provide, in part, an updated site-specific decommissioning cost estimate. If little decommissioning has been completed, and inflation and disposal costs have not changed, the cost estimate required by 10 CFR 50.82(a)(8)(iii) may be acceptable. NRC staff is not requiring the licensee to submit any contractual documents or agreements that exist between the licensee and the decommissioning contractor, and the cost estimate should not be impacted by the election of the licensee to decommission the facility, or contract to decommission the facility. However, for NRC staff to be able to make a finding that sufficient funding is available to complete decommissioning, the updated cost estimate of the remaining site dismantlement activities and the remediation plan that outlines how the decommissioning will be conducted must correlate. The updated cost estimate should be based on the remaining activities and the plans on how the actions will be completed. The updated site-specific cost estimate must address the remaining activities necessary to complete decommissioning, ensuringsufficient funds are available. Per 10 CFR 50.75, the financial assurance instrument must be funded to the amount of the cost estimate. During decommissioning the licensee may withdraw these funds to assist with decommissioning activities.

Lesson 8: Records

Old records may be inadequate or inaccurate for the purpose of developing the historical site assessment (HSA) and site characterization. The NRC staff suggests that these records not be relied on as the sole source of information for the HSA and site characterization although, in some cases, experience has shown that old records and results of operational surveys and post-shutdown scoping surveys have been submitted as substitutes for characterization surveys. For example, the results of operational surveys may represent the current radiological status, describing conditions over a limited time span, or may have been conducted to address specific events (i.e., post-spill cleanup assessment). In a few instances, the results of personnel interviews and information, which can only be considered as anecdotal, have been presented in the HSA. In fact, it could not be determined whether this information was part of an unbroken chronological history of the site or contained time gaps for which operational milestones or occurrences were missing. Although NRC staff encourages licensees to review old records and conduct personnel interviews (past and current employees and key contractors), there is a need to present this information in its proper context and qualify its usefulness and how it might be supplemented (e.g., via additional data searches or characterization surveys). To achieve the purpose of the HSA, a complete history of the residual contamination is needed. Given their importance, the NRC staff suggests that characterization surveys be developed only after the licensee has conducted a thorough evaluation of the information collected during the site historical assessment.

Based on the review of several DPs and LTPs, the NRC staff has found that licensees have generally done extensive characterizations of facilities slated for decommissioning. A review of selected characterization files (in support of decommissioning and turnover surveys) revealed that a wealth of information is indeed available, but that it is not conveyed or presented clearly in DPs and LTPs. The information NRC staff seeks can be drawn from existing characterization records or supplemental analysis of existing samples, thereby avoiding the need to conduct additional surveys and to send workers into radiation areas — all while minimizing costs. The type of information that is needed to support the preparation of DPs and LTPs focuses primarily on residual levels of contamination remaining on building surfaces or in soils (surface and subsurface), after the remediation work has been completed. The characterization of elevated contamination levels typically found in radiation areas is of no concern in addressing the design of FSSs, since these areas are contaminated at levels that obviously exceed any realistic $DCGL_W$. NRC staff is seeking a better presentation, and perhaps evaluation, of existing data supporting specific DQO elements and justification for the approach proposed in developing survey designs. In most instances, it is not a question of generating more data — rather, it is a question of making use of all existing data. There may be some exceptions where additional characterizations might be warranted. Such exceptions might apply to the characterization of subsurface soils, ground water, and TRU, since these may present unique challenges, but can be resolved without unnecessary radiation exposures.

Lesson 9: Environmental Reviews

In accordance with the provisions of the National Environmental Policy Act (i.e., Public Law 91-190), all agencies of the Federal Government are required to assess the environmental impact of any major Federal action that may significantly affect the quality of the human environment. As part of NRC's approval of either a DP or an LTP, NRC staff is required to determine if that approval is a Federal action. Therefore, the impacts on the human environment associated with NRC approving either a DP or an LTP must be assessed. Further, this assessment must include both radiological and non-radiological impacts. Although most licensees normally provide sufficient information for the NRC staff to assess the radiological impacts on the human environment, some licensees have not provided sufficient information related to current site-specific non-radiological impacts.

Because actions considered by the NRC when approving a DP are different than those associated with NRC's approval of an LTP, the information required to assess the impacts on the human environment are different. That is, when NRC approves a DP, NRC is approving the licensee performing the activities necessary to remediate radiological contamination at a site. Therefore, a DP should include information addressing non-radiological impacts on the human environment associated with these proposed activities. Non-radiological impacts include, but are not limited to the following: land use; water quality; transportation; air quality; ecological, historical, and cultural resources; hazardous material/waste; noise; visual/scenic quality; socioeconomics; and public and occupational health. However, under the provisions of 10 CFR 50.82, most if not all activities necessary to complete site remediation can be completed under the provision of 10 CFR 50.59. Therefore, these activities will not require prior NRC approval. Consequently, unless certain site-specific issues exist, NRC, when the NRC staff approves an LTP it is approving only (a) the adequacy of the decommissioning funding plan to assure that sufficient funding is available to complete the remaining radiological remediation activities, (b) the radiation-release criteria for license termination, and (c) the adequacy of the design of the final survey to verify that the release criteria have been met.

Lesson 10: Characterization Surveys and Classification of Survey Units

The NRC staff recommends that submittal of the DP or LTP occur only after sufficient site characterization has occurred. The NRC staff suggests that the DP or LTP 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 and LTPs have been submitted with incomplete or inadequate characterizations of radiological conditions. A review of such DPs or LTPs has shown that the lack of information makes it difficult to agree with the rationale justifying 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 either a DP or an LTP:

- *Use of operational, post-shutdown scoping, or turnover surveys as characterization surveys* — Characterization surveys are the most comprehensive of all surveys, yield the most information, provide the basis to design the FSSP, and are used for dose modeling as well. Characterization surveys are conducted to determine the current extent and magnitude, and variability (as surface and depth profiles) of the contamination, and radionuclide distributions and concentrations. Characterization survey results are used to guide remediation efforts, provide information with which to update waste volume and cost estimates, and develop DCGLs. Given their importance, the NRC staff recommends that characterization surveys be developed only after the licensee has conducted a thorough evaluation of the information collected during the HSA, and the results of operational surveys and post-shutdown scoping surveys. Accordingly, it is not appropriate to use the results of past operational and post-shutdown scoping surveys as substitutes for characterization surveys conducted using the guidance of MARSSIM. For example, the results of operational surveys may represent radiological status describing conditions over a brief operational time span or may have been conducted to address specific occurrences (i.e., post-spill cleanup assessment). Moreover, the results of both operational and post-shutdown scoping surveys may be of limited use unless it can be shown that data quality, instrument calibration methods, and detection sensitivities (fixed and scan measurements) for the anticipated radionuclide mix are comparable to those defined for the characterization surveys based on MARSSIM guidance. These limitations also apply to turnover surveys conducted after the completion of remediation. In all three instances, this approach is also a departure from the MARSSIM methodology in that it defeats the statistical basis intended to confirm that survey units meet the release criteria. As is noted in MARSSIM (Section 5.5.2.5), "Measurement locations based on professional judgment violate the assumption of unbiased measurements used to develop the statistical test described in Chapter 8" (of MARSSIM). If a licensee were to use turnover survey data for part of the final survey, statistical samples and/or measurements may need to be identified in addition to the turnover survey data. Also, the samples and/or measurements should be collected or made in compliance with MARSSIM guidance (i.e., random start and systematic sampling/measurements using an established grid) or other survey methods found acceptable to NRC staff.

- *Reclassification of Survey Units* — It may not always be appropriate to simply separate out an area of elevated activity as an individual Class 1 survey unit from a Class 2 or Class 3 survey unit since the initial basis for evaluating a Class 2 or 3 survey unit is based on specific criteria [i.e., 10 to 100 percent scan coverage for Class 2, and judgment (typically <10 percent) for Class 3 survey units]. Similarly, licensees should provide the basis in delineating Class 3 survey units as buffer zones around Class 1 and 2 survey units and areas with insufficient justification to be classified as non-impacted. If survey results were to reveal elevated levels of contamination in an arbitrarily selected portion of a Class 2 or 3 survey unit, then the classification of the entire survey unit should be deemed suspect and re-evaluated using MARSSIM guidance. In this context, the NRC staff suggests first, that there should be considerations of the assumptions made as to how the survey unit was initially classified, most likely or known causes of contamination, and the possibility that other similarly contaminated areas within the original survey unit might have gone undetected. The NRC staff also suggests that a DP or LTP address these considerations and describe the method, consistent with MARSSIM, that will be used if a survey unit or portion of a survey unit must be upgraded to a higher classification level. In general, increasing the coverage of the scan is less expensive than finding areas of elevated contamination levels later in the process. Finding areas with elevated levels of contamination later in the process will require the conduct of additional surveys, lead to delays in reconsidering the initial classification of the survey unit, and will lead to additional regulatory scrutiny. The NRC staff recognizes, in many instances, that DPs or LTPs are submitted at a time when some characterization work is still ongoing and that supplemental data may lead to the reclassification of some survey units. Accordingly, a DP or LTP should include the flexibility to accommodate changes in the classification of survey units as more characterization data are obtained and evaluated.

- *Completeness of Characterization Survey Design and Results* — In some submittals, the NRC staff has noted that contamination results for plant structures, systems, and components; surface and subsurface soils; and groundwater are at times incomplete. For example, the review of data characterizing such areas or media has revealed that only limited information is being provided about the presence of TRU (e.g., plutonium-239, americium-241) and hard-to-detect radionuclides (e.g., hydrogen-3, carbon-14, nickel-63). In other instances, the data fail to provide sufficient information in determining the fraction of surface radioactivity that is fixed and removable. Similar shortcomings were noted for removable alpha and beta radioactivity found in embedded piping, usually contained in residues, sediments, and internal film coatings. Although reporting histories of fuel cladding failures, some plants have not provided information on the presence of TRU in plant systems and at effluent discharge points. The characterization of neutron activation products in concrete and rebar is often limited in scope, and the presentation of the results fails to address the significance of the reported radionuclide concentrations and their applicability to other areas of the plant. In summarizing characterization results, there are instances when both the average and maximum surface beta activity results are below the stated MDCs. Such results are misleading since it is not clear if the stated MDCs are representative of all areas within a survey unit or whether there might be multiple MDCs that could be unique to distinct areas within each survey unit. Such results imply that the variability may apply to all areas within a survey unit, when perhaps the variability of the contamination might be multi-modal if it were evaluated by separate and smaller areas. This problem, in part, is attributed to how the data are edited for summarization. In other instances, licensees have proposed radiological results characterizing radionuclide distributions and concentrations using smears/wipes, air filters, and debris, with no rationale as to the relevance of the information. It should be noted that characterization survey results provide the most important information (i.e., the basis to design the FSSP; define radionuclide distributions and concentrations; identify hard-to-detect radionuclides and develop surrogate ratios; define survey area classifications; and assign the sigma characterizing the variability of the contamination (a key parameter in determining the number of samples in survey units)). Accordingly, the planning and execution of any characterization surveys should be conducted in a manner that will generate technically defensible results with which to design the FSSP.

Lesson 11: Embedded Piping

Nuclear power reactors and other types of nuclear facilities contain embedded piping that may become radiologically contaminated as a result of licensed operations. The NRC staff suggests that DPs and LTPs include a discussion on the methodology for conducting surveys of embedded piping planned to be left behind. The NRC staff suggests that sufficient justification for the assumptions considered in the computer modeling and dose analysis for embedded piping be described in the basis. Also, the NRC staff suggests that copies of relevant computer code printouts be included for NRC staff evaluation.

One approach that has been approved for surveys of embedded piping is to establish a separate site-specific dose criterion for external penetrating gamma radiation emitted from the internal surface of embedded piping present in structures (e.g., walls, floors, ceilings) which are also in the same survey unit. In this approach, the predominant radionuclide of concern from a dose perspective (e.g., cobalt-60) is determined by isotopic analysis of scale or residue samples collected within such piping during the licensee’s radiological characterization program. The dose criterion should be based on bounding conditions developed from characterization data, computer modeling using a radiation shielding computer code, and a detailed dose analysis of the exposure scenario. In the model, grit blasting of the internal surface of embedded piping may need to be considered to assess (a) any gains from the removal of loose surface activity and (b) whether the application of grout to immobilize and encapsulate fixed residual surface contamination would reduce radiation exposures.

It is important to describe the mechanism in which the dose contribution from the embedded piping and the non-embedded piping portion in a given survey unit is evaluated, when the dose to either component is determined to be equal to/or greater than the respective established dose limit, to ensure that the entire survey unit does not exceed the release criteria. Further, the NRC staff recommends that licensees discuss how adequate scan and static investigation levels will be implemented and further evaluated, as needed, in the FSS. It is also advisable that radiation detectors used for embedded piping surveys be properly calibrated for this specific geometry [including the use of National Institute of Standards and Technology (NIST) traceable radiation source(s)], which are appropriate for types, energies, and residual concentrations expected in the FSS.

Lesson 12: MDCs

The decommissioning process typically involves sites with multiple radionuclides present at the time the FSS is conducted. Although individual radionuclides and their respective $DCGL_W$ values and initial-scan MDCs for the principal radionuclides of concern have been identified, DPs and LTPs should describe the methodology and basis on which to implement a scan MDC to account for a mixture of radionuclides that may remain in a given survey area/unit. The NRC staff recommends that parameter values such as source (ε_s) and instrument (ε_i) efficiencies, surveyor efficiency (ρ), and performance criteria (d'), which determine the scan MDC, be evaluated before implementation; also, changes in the default parameter values (e.g., $\rho = 0.5$, $d' = 1.38$) need to be clearly justified in the DP or LTP.

In MARSSIM, decisions are made on selecting appropriate detection sensitivities or MDCs for radiological survey and laboratory instruments in the DQO process. Static MDCs within 10 to 50 percent of the $DCGL_W$ of the individual radionuclide are often readily achievable; however, the scan MDC involves a larger number of arbitrary assumptions and decisions. The NRC staff generally considers the ε_s values described in International Organization for Standardization (ISO) 7503–1 and ISO 7503–3 guidance for alpha- and beta-emitters to be acceptable estimates, absent site-specific information, for surface contamination detectors in the final status survey design. The NRC staff suggests that, in situations where surface contamination measurements are planned on irregular and uneven surfaces such as scabbled concrete and embedded piping, licensees determine an appropriate site-specific s value(s). Further, the NRC staff recommends that the methodology and basis for the ε_s value(s) be provided for NRC review.

When multiple radionuclides are present in the survey area/unit, application of an ε_i value, the use of a representative, conservative, or beta-weighted average energy for the anticipated radionuclide mixture, has been acceptable to NRC staff.

Because the estimated–scan MDCs for open land areas (soils) (Table 6.7 of MARSSIM) are premised on certain decisions and assumptions involving human factors and survey techniques, detector characteristics and performance, and computer modeling, it is advisable that licensees validate (e.g., *a posteriori*–scan MDC) the *a priori*–scan MDC used for design goals, as information is collected and assessed, so that an actual-scan MDC can be calculated for implementation in the FSS, for demonstration of compliance.

3.0 Lessons Learned During Decommissioning Final Status Survey In-Process

Inspections and Confirmatory Surveys

3.1 Introduction

Confirmatory surveys and in-process inspections conducted at various NRC materials and reactor licensee facilities undergoing decontamination and decommissioning have identified a number of issues with implementation of FSSes in accordance with current guidance. The issues identified are related to the following categories: instrumentation, procedural, and survey planning and data evaluation. Each issue is discussed together with the potential problems that may result and recommended solutions. It is important to note that this is not an all-encompassing discussion of identified issues, rather the discussions represent more recent and pervasive technical deficiencies.

3.2 Instrumentation

The issues discussed in the following sections relate to the selection and operation of either radiation detectors or instruments selected for radiological surveys.

3.2.1 Temperature Effects on Gas Proportional Detectors

Industry guidance for the calibration of instrument/detector combinations used for assessing residual radioactive material contamination levels requires calibrations be performed in a manner that simulates the environmental and set up conditions under which the equipment will be used (ANSI 1997 and NCRP 1991). Recent evaluations of surface activity discrepancies when evaluating comparative licensee and confirmatory survey measurements found a systematic under response in the licensee's reported activity levels during cold weather periods. In these cases, alpha plus beta or beta-only scans or measurements were being conducted using gas proportional detectors.

Normal practice is to conduct calibrations in a laboratory setting. When these instruments are distributed for use, the conditions may change once the user is out in the field. Temperature variations on the order of 30 to 40°F over the course of a work day are common. Past investigations have determined that the optimal operating voltage plateau, established during calibration, shifts while the instrument is being used under varying temperature conditions. For example, a controlled experiment showed for a specific brand and model of detector and instrument that a detector voltage plateau calibration at 70°F provided a plateau ranging from approximately 1700 to 1760 volts (optimal setting of 1725 to 1750 volts — the mid-point of the plateau). For the same detector calibrated at 20°F, the voltage plateau ranged from 1775 to 1875 volts, with an optimal setting of 1825 volts. The above data show that the voltage set point for the 70°F calibration is actually below the knee of the 20°F calibration. This shift — found to begin around 40°F — results in a detector calibrated at room temperature, then operated in a cold environment, to under respond to the source of radiation.

There are solutions to this issue to ensure calibration is conducted to match any expected temperature extremes. Voltage plateaus may be performed at multiple temperatures when possible. The proper voltage set point may then be selected to match conditions. Otherwise, to minimize the effects of the plateau shift during cold weather operation, select an operating voltage that is closer to 3/4 of plateau maximum. As conditions change, the high voltage may be adjusted to bring the detector back to within established operational parameters and verified through appropriate procedures. For hot weather, the opposite effect is probable and the operating voltage may have to be reduced to avoid a shift into the continuous discharge region of the plateau.

3.2.2 Instrument Count Rate Plateaus

Some types of data logging instrumentation selected for FSSes have audio response limitations that impact the surveyors ability, under certain conditions, to discern the presence of elevated activity. These instruments have a preset audio response that plateaus once the count rate reaches 4500 counts per minute (cpm). This condition does not impact alpha contamination assessment and normally does not interfere with assessing beta surface activity when ambient gamma backgrounds are at typical environmental levels. On the other hand, when these instruments are used for conducting gamma scans using NaI scintillation detectors, complicating factors occur. Typical background gamma levels range from approximately 2,500 to 12,000 cpm when using the more common NaI detector crystal sizes. It can be immediately seen that the background may saturate the audio capability of the instrument making it impossible for the surveyor to rely on increases in audio response to identify locations of elevated direct gamma radiation.

There are essentially three solutions to this problem, all of which result in either additional complicating factors that must be addressed or potential further project costs. One such approach that has been implemented is the use of an alarm set point action level — rather than relying on the audio response — that roughly correlates a specified count rate to a concentration in soil. The difficulty encountered with this approach occurs when the scan MDC calculations prescribed in MARSSIM are adapted. This is further discussed in Section O.3.3.1 of this appendix. Furthermore, the use of such an action level should build-in adequate conservatism — and corresponding confidence level — to account for the statistical variance normally seen in the data that are used to generate the count rate to pCi/g relationship. Another option is to use the audio divide feature of the instrument to bring the audio response below a suitable fraction of the plateau. When doing so, the survey planner will need to ensure that the reduced audible background count rate is factored into the MDC calculations. Lastly, consideration may be given to using a different instrument or smaller NaI crystal size (to lower the background) for gamma scanning.

3.2.3 Miscellaneous Instrument Issues

Other instrument issues that have been noted include static (disconnection from a continuous gas supply) operation of gas proportional detectors, long detector to instrument cables, and altitude effects on the calibration of gas proportional detectors. When gas proportional detectors are operated in a static mode, there will be some gas leakage from the detector. As the gas supply decreases, the detector efficiency degrades accordingly. The rate of gas leakage greatly varies among detectors, particularly once the factory-installed face and gasket are removed for maintenance. The rate of leakage has been observed to range from minutes to days. Past field observations of FSSes and comparative measurements have found that these detectors may have had only a partial purge, resulting in the underestimation of surface activity levels. Therefore, procedures should specify that when surveying in a static mode, the operational parameters should be checked regularly through either a background or source check. If the detector falls below established parameters, repurging the detector would be required prior to continuing surveys. Operation at the alpha plus beta voltages more readily allows the surveyor to distinguish a drop in efficiency caused by gas leakage as the background levels — generally in the 200 to 500 cpm range for hand-held detectors — will noticeably decrease. However with the 0 to 5 cpm alpha voltage backgrounds of most hand-held gas proportional detectors, a decrease in efficiency will not be immediately observable and therefore will necessitate a regular operational source check to validate performance.

Section O.3.2.1of this appendix discusses the importance of calibrating instruments under the same environmental and set up conditions in which they will be used. Two additional factors that have occurred are differences in detector performance that result from significant changes in altitude between the calibration and use point and when long cables are used. A gas proportional detector calibrated at 1000 feet above mean sea level will under respond when operated at higher altitudes. Therefore, this impact must be addressed by either calibrating at the site where the equipment will be used or otherwise adjusting the electronics once the equipment is received at the site to ensure operational parameters are correct. Similarly, there have been cases where the original 5- or 6-foot cable that an instrument/detector combination was calibrated with is replaced with a longer cable to permit access of the detector to difficult to reach places. The longer cable may increase the electrical impedance and again result in an under response. The instrument/detector combinations should then have separate calibrations performed, both with the standard and long cables.

3.3 Procedural

The issues discussed in this section were identified either as a result of observation of FSSes during in-process inspections or following the review of licensee procedures.

3.3.1 Alarm Set-Points and the MARSSIM Scanning MDC Calculation

There have been a number of instances where FSS procedures have implemented the use of various detectors coupled to data logging instruments. These instruments in several cases were set to alarm at a pre-determined count rate action level that is calculated to correspond to the $DCGL_W$, rather than relying on the surveyor listening to the audible response. Although this may be an acceptable practice, with the provision of an adequate technical basis, the MARSSIM scan MDC equations are no longer appropriate. The reason for this position is that the derivation of the scan MDC equations are based on signal detection theory. That is, how a human observer theoretically processes the audible input and then makes decisions. Refer also to Section O.3.3.3 for related discussions.

Furthermore, where a human may continually adjust to varying backgrounds, an alarm set point is normally established as a multiple of a static background. However, once an alarm (or MDC) is set using a static background, the electronics are not capable of discriminating when lower or higher background areas are encountered. As a result, any significant changes in background levels would necessitate a re-evaluation of the basis of the MDC in determining whether the new MDC still meets the related data quality objectives. In the case of operating in a lower background area, the instrument may not alarm when required.

The following example illustrates this point: Background is established at 10,000 cpm. The action level is determined to correlate to 5,000 net cpm above the selected background, or 15,000 gross cpm. However, backgrounds fluctuate between 8,000 and 11,000 cpm in the survey unit, dependent upon surface types. While surveying in an area where the background is 8,000 cpm, residual contamination contributing an additional net count rate of 5,000 cpm would fail to activate the alarm (13,000 gross cpm) — increasing the false negative rate. Conversely, when operating in an area where background is higher than the set point background, one would expect a higher false positive rate.

The use of an alarm set point therefore requires a number of considerations for calculating the scan MDC, procedures for addressing varying backgrounds, and specific investigation requirements for when an alarm occurs (i.e., second stage scanning and soil sampling).

3.3.2 Gamma Fixed Point Measurement in Place of Surface Scanning

Confirmatory surveys conducted in Class 1 soil survey units at several sites have identified small areas of residual gamma-emitting contamination that when evaluated, exceeded the $DCGL_{EMC}$. A root cause analysis was performed and determined that the site procedures required systematically spaced, fixed point gamma measurements rather than prescribing surface scanning over 100 percent of the survey unit area in accordance with MARSSIM.

Experience has shown that for characterization surveys, where contamination may be more distributed, systematic fixed point gamma measurement can be useful for identifying large areas requiring investigation. However, once an area is remediated, contamination generally becomes more isolated and heterogeneously distributed. Scanning surveys are designed to specifically address this condition where small areas of elevated activity may be present that would go undetected by systematically-spaced measurements. Therefore, to resolve this issue, surface scans should be performed over, not only Class 1, but all survey units following the MARSSIM recommendations for coverage.

3.3.3 Not Listening to Audio Response While Conducting Surface Scans

A significant number of facilities assessed during decontamination and decommissioning do not require the surveyor to listen to the instrument audio response while conducting radiological surface scans. Rather, the analog meter is visually observed, an instrument alarm is set to notify the surveyor when to pause and investigate, a peak trap mode (the maximum observed count rate value is stored in the instrument memory) is used and the data are reviewed for anomalies post-survey, or a second person — rather than the individual using the detector — listens to the instrument audio.

Each of these techniques have inherent deficiencies that impact one's ability to identify locations of residual contamination. The instrument alarm comments were detailed in Section O.3.3.1 of this appendix, with one additional comment provided here. That is, it has been previously observed during in-process inspections that a only a single alarm may occur when multiple hot spots were known to be present or, if a peak trap mode is used to assess scan data, only the maximum value is available for review. With both of these approaches, information on the presence of multiple areas of elevated direct radiation is not available to the surveyor.

Reviews of procedures and direct observations of FSS field scanning techniques identified a unique variation of this issue. In these cases, a surveyor separate from the one performing the survey listens to the audio output of the instrument. The previous discussion of the applicability of the MARSSIM recommended scan MDC calculation also should be considered when using the dual surveyor approach. A second key component that should be addressed is the impact on second stage scanning. That is, the mechanism for when the surveyor moving the detector is caused to stop the detector and investigate an increase in the count rate.

Lastly, there are two less common methods that have been used during FSS scans. The first is reliance on visually observing fluctuations in the instrument readout — either needle deflections or digital readout. Again, this is contrary to the MARSSIM scan MDC paradigm and this method is significantly less sensitive than the audio output due to instrument smoothing functions built into the readout, may require a greater degree of vigilance, and also may result in additional safety concerns. The second method is the use of ratemeter-scalers capable of counting alpha and beta interactions simultaneously. These instruments provide a different tone for alpha or beta counts. Although in one specific case, the surveyors were listening to the audio response, it was found during confirmatory surveys that a significant quantity of alpha contamination had not been identified. The most probable cause was the difficulty in discerning the low alpha activity guideline over the higher beta background. In other words, the beta activity count rates overwhelmed the surveyors ability to audibly detect the low alpha count rates that required further investigation. It is therefore recommended that separate alpha and beta scans be performed.

For any of these cases, the surveyors should listen, using head phones — especially in high noise environments — to the audio output. The use of the other techniques described above do not adhere to the MARSSIM guidance and therefore may require preparation of a technical basis that details the approach, calculated scan MDCs for the specific approach, procedures for second stage scanning and investigation requirements.

3.3.4 Instrument Calibration for Assessing Surface Activity Using ISO 7503-1

The implementation of the instrument calibration guidance for assessing alpha and beta surface activity recommended in ISO 7503-1 (ISO 1988) and adapted into the MARSSIM is not always consistently applied. This issue was identified while reviewing either the LTP or specific licensee calibration procedures. The ISO 7503-1 guidance more accurately accounts for surface conditions encountered at decommissioning sites — typically rough, dirty, or porous — and emission energy of the radionuclides of concern. Without the proper application of the ISO 7503–1 guidance, surface activity levels for alpha and low-energy beta-emitting contaminants will be significantly underestimated. The guidance recommends a total efficiency that is the product of two components — an instrument efficiency (ε_i) and a source efficiency (ε_s).

The most commonly encountered calibration findings have identified the use of a 4π total efficiency instead of the ISO 7503-1 and MARSSIM–adapted 2π instrument efficiency which is then modified to address surface conditions ($\varepsilon_i \times \varepsilon_s$). As an example, if technicium-99 — a low energy beta emitter — were the contaminant of concern at a site, an expected laboratory derived 4π efficiency for a Geiger-Mueller (GM) detector would be approximately 0.17. However, this efficiency is overly optimistic because of the expected attenuating surfaces that will be measured in the field. The comparative ISO 7503-1 derived technicium-99 efficiency would be approximately 0.05 (0.20 for the ε_i × 0.25 for the ε_s) for the same GM detector. The resultant surface activity would be underestimated by a factor of almost 70 percent using the 4π efficiency versus the two component ($\varepsilon_i \times \varepsilon_s$) efficiency. A related issue identified that also results in an underestimation of residual contamination was the application of an ε_s for alpha calibrations of 0.5 — the correct default value is 0.25 for alpha emitters and low energy (< 400 keV maximum energy) beta emitters.

In general it has been seen that licensees have adequately accounted for mixtures of varying energy beta emitters, hard-to-detect radionuclides, and unusual surface configurations such as corrugated metal in determining total efficiencies.

3.3.5 Performing Alpha Rather than Beta Surface Activity Measurements for Natural Thorium Surface Contamination

There have been several instances where residual natural thorium surface contamination was assessed by performing only alpha activity measurements. Natural thorium emits both alpha and beta radiations, therefore, either alpha or beta activity may be measured for determining the residual activity of the thorium contaminant. However, beta measurements provide a more accurate evaluation of thorium contamination on structural surfaces due to the problems inherent in measuring alpha contamination on rough, porous, and/or dirty surfaces. For the thorium series in secular equilibrium, for each beta emission there are approximately 1.5 alpha emissions — a beta to alpha ratio of 0.67. At one site, both alpha and beta surface activity measurements were performed during confirmatory surveys at the same location and the results compared. The data clearly showed the significant and widely varying alpha attenuation with beta to alpha ratios ranging from 3 to 280 — much greater than the theoretical ratio of 0.67. This provides further evidence that alpha activity is difficult to measure on surfaces that are typically encountered during radiological surveys and when possible, beta measurements should be performed. Alternatively, the alpha efficiency should be empirically reduced to account for the attenuation.

Uranium contamination on surfaces presents similar challenges as natural thorium when planning for the type of surface activity assessments that will be performed. As with thorium, the uranium series also emits both alpha and beta radiations. The specific alpha to beta ratio for the type of uranium (natural, natural processed, enriched or depleted) should be determined. Dependent upon the uranium isotopic abundances, these alpha to beta ratios can range from approximately 1:1.6 for depleted uranium, up to 20:1 for highly enriched uranium. This information is necessary for selecting which emission to measure, calculating an appropriate efficiency, and quantifying surface activity.

Several sites where natural thorium was the contaminant measured the alpha component rather than the beta component due to high ambient gamma background levels. This approach was followed because the high ambient gamma background present at the site resulted in a static beta surface activity measurement MDC that exceeded the thorium surface activity guideline. Alpha measurements were therefore selected to demonstrate compliance. However, the significant alpha attenuation was not accounted in the detector calibration. As a result, the reported alpha surface activity significantly underestimated the residual thorium contamination.

The impact of the high ambient background can be readily resolved by revising procedures and adapting one of two methods. If a given approach must rely solely on alpha measurements to assess residual thorium (or uranium) activity, alpha calibrations should then be conducted in accordance with the ISO 7503-1 and MARSSIM guidance. That is, the total efficiency of the detectors should be modified to account for the significant source attenuation. A second approach, would be to conduct beta activity measurements corrected for the high ambient gamma background. This is accomplished by performing both shielded (using a sufficiently thick Plexiglas™ shield) and unshielded measurements both in the survey unit and a suitable reference area. The surface activity:

$$dpm / 100cm^2 = \frac{Net\ count\ rate\ (N)}{\varepsilon_{tot} \times Geometry \times other\ \mathrm{mod}ifying\ factors} \qquad \text{(O-1)}$$

is calculated with correction for the gamma component. The net count rate used in the numerator of the surface activity equation is acquired as follows:

$$N = \left(R_{u,su} - R_{s,su}\right) - R_{rm} \qquad \text{(O-2)}$$

where N = net counts
$R_{u,\,su}$ = unshielded survey unit count rate
$R_{s,\,su}$ = shielded survey unit count rate
R_{rm} = reference material background count rate (ambient background subtracted out)

$$R_{rm} = R_u - R_s \tag{O-3}$$

where R_u = unshielded (gross) on background reference material and
R_s = shielded, background count rate.

Example: Beta activity measurements are required on a survey unit concrete floor. There are high ambient gamma levels in the survey unit due to contaminated sub-floor soils. A non-impacted concrete floor in another part of the facility is identified for background reference measurements. The count times are for one minute, the ε_{tot} is 0.20, and the geometry factor is 1.26. The following background reference material data for the concrete floor are obtained:
R_u = 400 cpm; R_s = 300 cpm (the gamma component of the background).
R_{rm} = 400 cpm – 300 cpm = 100 cpm.

The following survey unit concrete floor data are obtained: $R_{u,\,su}$ = 1000 cpm; $R_{s,\,su}$ = 500 cpm. Therefore, N = (1000 cpm – 500 cpm) – 100 cpm = 400 cpm. When this value and the previously provided count time, geometry, and total efficiency are substituted into the surface activity equation, the reported surface activity result equals approximately 1,600 dpm/100 cm^2.O.3.4

3.4 Survey Planning and Data Evaluation

The following sections describe issues encountered that are related to survey planning input parameters and subsequent post-survey data evaluation.

3.4.1 Contaminant Variability Ratio: Difference Across a Site

There have been several instances where a limited number of soil samples were used to determine a site-wide ratio between various contaminants. A surrogate contaminant was then to be measured and the ratio used to account for the remaining site contaminants. In one case, the sampling procedure did not take into account the actual site spatial contaminant distribution. Instead, a limited sample data set from one area of the site was relied upon to prepare the radionuclide ratios. A review of site data collected during earlier scoping surveys clearly demonstrated that the ratio varied among the radionuclides of concern, dependent upon which area of the site the sample represented. When the varying ratios were analyzed, it was determined that the site-specific surrogate ratio that had been developed would significantly underestimate the inferred radionuclide concentrations for portions of the site.

This issue can be readily avoided provided representative samples are collected in such a manner that the ratio developed accurately represents both spatial, and in some cases, depth variability. Furthermore, it may not be reasonable to select a single ratio for application across a site. Rather, it may be necessary to develop multiple ratios and specifically identify sites areas where each ratio will apply. In other cases, the ratio may vary to the extent that no consistent ratio can be inferred, meaning the surrogate approach would not be an option and radionuclide-specific measurements are then required. Additionally, the ratio is typically verified for a percentage of the FSS samples. This is especially true in remediated areas where the decontamination may alter the ratio through either physical or chemical processes.

3.4.2 Unity Rule Not Used with Multiple Contaminants

Recent reviews of FSS data packages have identified a critical oversight with demonstrating compliance with the release criteria at some sites with multiple contaminants. What has occurred is that each individual radionuclide is compared with the respective $DCGL_W$ and a conclusion reached as to the acceptability of a survey unit for release. However, an additional requirement is to apply the unity rule (also known as "sum of fractions") to the data to ensure that the basic dose limit is met. This is based on the $DCGL_W$ for each radionuclide equating to the dose limit for release of the site. Due to the additive nature of the dose from each radionuclide, the total residual activity must be proportionality reduced to ensure the sum of each radionuclide divided by its $DCGL_W$ does not exceed one (unity). Application of the unity rule is detailed in Section 2.7 of NUREG-1757, Volume 2. Licensees should ensure that when multiple radionuclides of concern are present that the unity rule is applied both in data evaluation and in the initial survey planning phase.

3.4.3 Survey Unit Misclassification

Evaluations of licensee survey unit designations and confirmatory surveys have identified inconsistencies with recommendations on survey unit classification; primarily involving contaminated Class 2 survey units. That is, contamination in excess of the $DCGL_W$ that has been found during past confirmatory surveys within Class 2 survey units. As expected, the contamination was usually identified in that portion of the survey unit bordering adjacent Class 1 areas. The simplest solution for the observed occurrences would have been for the licensee to have extended the size of the Class 1 survey units to include adjacent regions. In one case, the contamination was found on the wall portion of the interface between the Class 1 floor and Class 2 wall.

3.4.4 Demonstrating Compliance with Hot Spots Present in a Survey Unit

There have been isolated instances where reviews of FSS data packages or confirmatory survey findings identified survey units where the $DCGL_W$ was statistically satisfied, but hot spots were not fully addressed. When hot spots remain in a survey unit, MARSSIM recommends additional data assessment to ensure compliance with the basic dose limit. The first recommendation is that each hot spot be evaluated against the $DCGL_{EMC}$, relative to hot spot size and allowable concentration within the hot spot area. Generally, for hot spots documented in FSS packages, this recommendation is addressed adequately. A component for demonstrating compliance that has been overlooked is showing that the combination of residual hot spot contamination in addition to any uniformly distributed activity is less than the basic dose limit. MARSSIM, Section 8.5.2, provides the equation and narrative guidance for implementation and documentation in survey units where this condition exists.

3.5 Implications of Contamination Identified During Confirmatory Surveys

The question is frequently asked: What should be done when contamination is identified during the confirmatory process? This question is directly pertinent because many of the lessons learned presented here have been identified as the root cause for missed contamination. There is no single answer to the question, as each situation is unique. The DQO process should be followed to establish remedies to the given situation. For confirmatory survey results contrary to the FSS reports for the site, the NRC staff and the licensee should determine what is the magnitude of the finding (number of anomalies identified, size of the anomalies, classification of the area where they were identified) and the proposed remedy. Anomalies that are identified should be evaluated for compliance with the $DCGL_W$ and $DCGL_{EMC}$ and a determination made if the area affected is acceptable relative to size and concentration, has the licensee previously documented and adequately addressed the anomalies, and are they within the bounds of survey unit classification? For multiple anomalies, determine the root cause and reevaluate DQOs. Also consider what percent of a site was subjected to confirmatory surveys. If a small percentage was investigated and multiple areas of residual contamination were identified, the confidence level that the FSS procedures were adequate and that the remaining site areas are acceptable would be low and may necessitate further licensee activities to remedy the data gaps. On the other hand, when a large percentage is confirmed and few anomalies are found, the confidence interval increases significantly and further activities to provide added assurance of guideline compliance may be minimal.

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