MDNR Response to Specifically Identified RAIs

RAI#1 In section 8-0, "Planned Decommissioning Activities", you state, in part, that planned decommissioning activities for the site include the removal of above-grade components of the Leachate Collection and Treatment System (LCTS). Other activities include cutting, capping, and sealing the LCTS piping just below grade level, and removing the LCTS Building. These activities would permanently disable the LCTS system. The Michigan Department of Environmental Quality (MDEQ), who has regulatory jurisdiction over the non-radiological hazardous materials on MDNR and S. C. Holdings sites, has indicated in conversations with the NRC, MDNR, and S.C. Holdings that the LCTS needs to remain operational and should not be disabled. The MDEQ believes that the system will be needed to remove leachate from waste cells at both sites to reduce the potential for leaks. We are concerned that if a new leachate system needs to be constructed in the future, thorium contamination from within the cell could be released and contaminate the cell cap and the surrounding environment. Provide a commitment to leave the LCTS in place or otherwise provide assurance that at some future date, MDNR will not be required to construct a new leachate collection system in the MDNR waste cell.

Requested Additional Information: MDNR acknowledges MDEQ's desire to retain the LCTS piping in place. MDNR further commits to leaving the in-ground LCTS piping in place within the MDNR cell provided that: the NRC agrees and accepts that future operation of the LCTS system for managing leachate migration from within the cell to the surrounding environment does not pose an unacceptable radiation dose to persons who might be exposed to radiation associated with such operation in the future (See answer to RAIs #3 and #4).

RAI#2 With regard to the waste cell, it is conceivable that if the LCTS is disabled, sufficient hydraulic pressure could build within the cell causing leakage of leachate. Provide additional information to ensure that disabling the LCTS System will not lead to leakage of leachate contaminated with thorium and/or its daughter products.

Requested Additional Information: First, MNDR makes no claim as to the absolute hydraulic integrity of the slurry wall system enclosing the MDNR waste cell. In fact, the slurry wall system is unrelated to the radioactive slag that is co-deposited there, although it does circumscribe the identified slag deposits. The slurry wall system was installed by Waste Management, Inc. in response to MDEQ directives related to the control and management of non-radiological hazards present at the site.

In making the decision to abandon and decommission the LCTS piping as described in the Decommissioning Plan, MDNR was addressing an NRC stipulated prohibition on the operation of the existing LCTS in its existing, NRC issued, radioactive materials license (SUC-1581). The decision to disable the LCTS system in the decommissioning process was made in consideration of the radiological implications of leachate leaking from the cell. Two distinct features of the proposed Decommissioning Plan were tailored to address this conceivable eventuality.

First, the conceptual site model used to assess the protectiveness of the proposed subsurface soil DCGL does not rely upon the integrity of the cell's slurry walls as a containment system. In fact, the conceptual site model does not take into account the presence of slurry walls, nor does it derive any advantage from the slurry wall's ability to retard the lateral movement of leachate. As a result, the dose attributable to recreational land use exposures to leachate containing residual thorium radioactivity (and/or its daughter products) that "leaks" past the slurry walls is accounted for in the conceptual site model and is protective.

Second, MDNR undertook a leachate sampling program to assess the radiological impact of the subsurface deposits of thorium-bearing slag on the radiological condition of the leachate. The data obtained from leachate sampling program (See the DP, Revision 1, Appendix I) provides solid evidence in support of the conclusion that the slag is highly insoluble and does not readily leach radioactivity into the fluid pore spaces within the cell. The majority of all sample results were found to be well below minimum detectable concentrations. In the few samples where radiological species were detected, radionuclides were present in concentrations consistent with those present in background. This data suggests that even if the slurry walls of the cell were to leak, the leachate itself is not impacted with elevated concentrations of residual radioactivity negating concern over slurry wall leakage.

While MDNR can make no assurance that "disabling the LCTS System will not lead to leakage of leachate" from the cell, it is clear from the fate and transport dose modeling that the radiological impacts to receptor dose are minimal and well within the 25mrem/y standard. It is also clear from the leachate testing performed that radiological impacts to the leachate are minimal (indistinguishable from background). Thus, MDNR maintains

that leakage of leachate from the cell, if it were to occur, is inconsequential from a radiological perspective.

RAI#3 If the DP is amended and the LCTS is not dismantled and remains operational, it is possible that during operation of the system, piping or tanks could leak leachate. If that leachate contains thorium or its daughter products, the cell cap and the surrounding areas could become contaminated. MDNR needs to identify the actions it would take if, during operation, the LCTS leaks leachate that contains thorium or it daughter products. Since MDNR requests unrestricted release of the site after remediation, the staff requires assurance that any future operation of the LCTS will not pose an unacceptable radiological dose to those who may be exposed to the leachate. Additionally, this dose needs to be identified in your response.

Requested Additional Information: Since MDNR had anticipated that the LCTS system would be decommissioned as part of the license termination process, no scenario involving "operational leakage" was previously evaluated to quantify the potential radiation dose to those that might be exposed to radioactivity stemming from leaks in the LCTS system should it remain in place and become operational. As described in its response to RAI #2, MDNR is convinced that the leachate is not adversely impacted with elevated concentrations of residual radioactivity associated with currently licensed materials. Nonetheless, MDNR understands the need to demonstrate that future operation of the LCTS (should it ultimately be left in place) will not pose an unacceptable radiological dose to those who may be exposed to the leachate.

MDNR has constructed an additional scenario (based upon the exposure parameters associated with the composite recreational user scenario) that simulates a "system leakage" setting in which the cap and the surrounding environs might become impacted and which bounds the exposure conditions that might reasonably exist under such a condition. The scenario involves revising the composite recreational user model such that leachate (near surface groundwater) is extracted and used to "irrigate" the site, thus simulating a spill or leakage condition. The analysis of the radiological impacts from this scenario corroborate MDNR's position that the leachate is not now radioactive and will not likely become radioactive through subsequent leaching from the entombed slag. In fact the most likely annual dose (50th percentile) as well as the 95th percentile estimate of annual radiation dose associated with this scenario, are both well below 1 mrem even when it assumed that thorium exists at its specific activity limit. The results of this analysis are attached (Attachment 1) for the NRC's review and consideration.

MDNR has requested approval of its decommissioning plan with the stated objective of license termination without restriction. The implication of license termination *without restriction*, as we understand it, is that the proposed remedy must be protective without reliance on some future action. Given that MDNR has herein demonstrated that "operational leakage" of the LCTS system does not result in annual radiation dose in excess of 25 mrem/y (the criteria under which unrestricted release is permissible), MDNR believes that actions intended to mitigate the radiological consequences of future LCTS leakage are not required or permitted.

RAI#4 If the LCTS remains operational, future workers may be exposed to radioactive contamination within the LCTS piping, wells, and tanks. MDNR should identify the potential dose to LCTS workers and visitors during operation of the system.

Requested Additional Information: Based on the existing leachate sampling data for radioactivity and the projected potential concentrations of radioactivity in leachate (as determined by RESRAD modeling), MDNR is convinced that future workers or visitors will not be exposed to appreciable amounts of radioactive contamination stemming from operations involving leachate handling. To further substantiate this position, and to comply with the NRC's request that MDNR "identify the potential dose to LCTS workers and visitors during operation," a radiation dose assessment is provided.

The potential future dose arising from "contact" exposure pathways (i.e., ingestion, inhalation, etc.) associated with future operation of the LCTS result from incidental contact with the leachate. The potential radiation dose from a worker's or visitor's *incidental contact* with the leachate is reasonably bounded by the "system leakage" scenario described in response to RAI #3. In that scenario, the only "contact" pathway that results in a potential radiation dose is the consumption of aquatic foods, an activity not likely to be engaged in by workers or visitors to a site involved in landfill leachate collection operations. Consequently, MDNR concludes that there is insignificant potential for radiation exposure via the "contact" pathways.

The unique properties of gamma radiation lead to an additional exposure pathway that does not require direct human contact with the leachate. The penetrating gamma radiation pathway (external pathway) could theoretically result in exposures to workers or visitors even when no leakage or direct contact with the leachate occurs. In order for this pathway to be complete in a practical sense, one must consider a setting in which above ground piping and tanks containing leachate are present. RESRAD is not suited for this type of exposure modeling. To assess the potential penetrating gamma radiation dose from above ground LCTS components, MDNR used the MicroShield computer modeling code (Version 6.02, Grove Engineering 2003). MicroShield is a photon transport modeling code specifically designed to assess the radiological exposure conditions resulting from a gamma radiation source.

MDNR again chose to perform a conservative bounding calculation to assess the potential future dose requested by the NRC. While it is considered an unlikely prospect, based upon conversations with representatives of S.C. Holdings, MDNR's dose assessment assumes that a large leachate-holding tank will be installed at the site. (Note: S.C. Holdings indicated to MDNR that their plan is to pipe leachate extracted from the MDNR cell directly to a single, common leachate collection/processing facility located at on the Waste Management Site property.) The assumption that a large leachate-holding tank will be installed at the MDNR site is bounding in that it represents a potential source term far larger than would be possible with above ground piping or other smaller vessels and containers that might be envisioned.

The conceptual tank containing leachate from the MDNR cell is assumed to have a large capacity (≈ 10,000 liters or 2,500 gal.) and is modeled completely full of leachate. The

concentration of residual radioactivity assumed present in the leachate is taken from the maximum concentration projected over the 1000-year outlook by RESRAD using the "system leakage" scenario. The "system leakage" scenario itself is exceptionally conservative in that it assumes the presence of thorium radioactivity in slag at the specific activity limit.

The MicroShield code projects a maximum potential gamma radiation exposure rate at a distance of 1 meter from the tank on the order of 10^{-6} mrem/h. The results of this analysis are attached (Attachment 2) for the NRC's review and consideration.

Assuming a worker spent an entire work year in immediate proximity to an operating LCTS system under prohibitively unlikely radiological conditions, the resulting annual gamma radiation dose would be well less than 1 mrem/y.

Additional Information, Not Specifically Requested:

As described in our responses to the NRC's RAIs above, MDNR anticipated and planned for the retirement of the LCTS piping and the removal of the LCTS building including its miscellaneous concrete appurtenances and the former decontamination pad as part of the site decommissioning activities (See Section 8.0 of the DP). However, as MDNR considered the responses to the NRC's RAIs concerning the possibility that the LCTS piping system could be left in place, it became apparent that LCTS building itself along with the former decontamination pad might also be left in place at the time of license termination[1].

MDNR requested input from the MDEQ and S.C. Holdings as to the perceived future need for the LCTS building and the former decontamination pad. S.C. holdings expressed no desire for the building or decontamination pad to be left in place, but the MDEQ insisted that these remain in place at the site. While inconsistent with MDNR's planned decommissioning activities and desire to restore the natural look of the site to the extent practicable, leaving these structures in place is not a significant radiological decommissioning issue given that it is deemed acceptable to leave the LCTS piping intact. Therefore, MDNR is proposing that the DP for the Tobico Marsh SGA Site be revised to allow for the LCTS building, its miscellaneous appurtenances, and the former decontamination pad to be left in place at the time of license termination to accommodate the wishes of other stakeholders.

The greatest impact on the MDNR's decommissioning plan arising from the decision to leave these structural components in place, is that the radiological criteria used to demonstrate compliance for a structure removed as part of the decommissioning process differs from the radiological criteria governing structures that will remain at the site at the time of license termination. When the building and other structures were to be razed prior to the request for license termination, the MDNR had proposed to perform materials and equipment release surveys on these structures using the surface contamination limits (PGD 83-23) approved in its NRC license. Structures that remain in place at the time of license termination are, instead, subject to the NRC's dose-based decommissioning criteria (10 CFR 20, Subpart E). To demonstrate compliance with the dose-based criterion, it was necessary to perform additional dose modeling to derive a "building surfaces" DCGL for those structures that may remain in place at the time of license termination. In addition to deriving dose-based DCGLs, MARSSIM compliant surveys have to be designed and references to the planned removal of the LCTS piping, building, and other miscellaneous structures in the DP have to be revised. The following text sequentially identifies those locations within the DP (Revision 1) where changes are proposed to address to the aforementioned revisions:

1 The MDNR has no need to retain the LCTS system piping, the building or any other structural features of the site in order to demonstrate compliance with the NRC's decommissioning rule and to terminate its radioactive materials license. In fact, the LCTS system was designed and installed by the MDEQ to manage leachate in the cell because of its non-radiological hazardous constituents (The leachate itself exhibits no enhanced radiological properties). However, the MDEQ and S.C. Holdings (a responsible party to non-radiological constituents at the site) objected to MDNR's plan to permanently disable the LCTS system in spite of the fact that it had never been used since being installed several years ago. MDEQ further objected to the dismantling of the LCTS building and its appurtenances as well as the former decontamination pad.

Section 1.5, Page 1-7, 2nd ¶. Replace the second paragraph and subsequent bullets with the following:

Since the LCTS building, miscellaneous concrete pads, and the above-grade appurtenances of the LCTS itself are to be left in place as components of the final condition for the site, they are subject to the decommissioning dose limit. Surface radioactivity DCGLs designed to satisfy the decommissioning dose limit have been developed for these site features. MDNR, in consultation with the NRC, chose to use the NRC's DandD computer-based dose-modeling code to derive the applicable DCGLs to be used when performing final status radiological surveys of the LCTS building and other structural components scheduled to remain in place at the time of license termination. The DandD code consistently produces conservative estimates of the applicable DCGLs when default parameter assumptions are employed. MDNR chose to derive DCGLs using the DandD code's default parameters, except that the exposure duration time was adjusted to more realistically depict the expected exposure conditions for the site. The derived surface activity limits corresponding to an annual dose of 25 mrem/y for the radionuclides present at the site are:

- 3,209 dpm/100 cm^2 (total alpha activity, averaged over the survey unit)
- 1,432 dpm/100 cm^2 (total beta activity, averaged over the survey unit)

Section 1.7, Page 1-8. Replace Section 1.7 with the following:

MDNR anticipates that the decommissioning project activities will be completed on December 30, 2005 with the submittal of the Final Status Survey (FSS) Report and request for license termination. Additional details related to project schedules are discussed in Section 8.5.

Section 5.1, Page 5-3. The current text discusses the use of the RESRAD modeling tool with regard to development of both the subsurface and surface soil DCGLs. Because the LCTS building, decontamination pad, etc. were slated for removal, this section offers no discussion of the dose modeling performed using the NRC's DandD code to derive DCGLs for the building surfaces. The following discussion supplements that found in Section 5.1:

After consultation with the NRC, MDNR chose to use the NRC's DandD computer-based dose-modeling code, Version 2.1.0 to derive surface activity DCGLs applicable to the LCTS building and other miscellaneous concrete structures that are to be left in place at the time of license termination. Historical information and routine radiological surveys provide a solid basis to regard the potential for building and other surfaces to be radiologically contaminated as very low. The DandD code consistently produces conservative estimates of the applicable DCGLs when default parameter assumptions are employed. MDNR chose to derive DCGLs using the DandD code's default parameters, except that the "Time in Building" parameter (To) was adjusted to more realistically depict the expected exposure conditions for the site. The "time in building" parameter was modeled as a constant (the default) with a value of 0.462 hours per week. The value

selected corresponds to an exposure within the LCTS building at a rate of 10 days per year and approximately 2.5 hours per day. Based on anticipated use of the LCTS building in the future, MDNR judges this assumption appreciably conservative.

The only other site-specific inputs to the DandD model are those describing the source term itself. The same isotopic composition described and used for the RESRAD modeling was also used to derive the surface activity DCGLs. The area of the contaminated surface was modeled as 22.3 m^2. Resulting estimates of annual dose equivalent were found to be insensitive to variability in area of contaminated surface parameter.

- Pb-210 0.5% ..... 11 dpm/100 cm^2
- Ra-226 1.1% ..... 23 dpm/100 cm^2
- Ra-228 16.1% ..... 341 dpm/100 cm^2
- Th-228 16.1% ..... 341 dpm/100 cm^2
- Th-230 50.0% ..... 1,060 dpm/100 cm^2
- Th-232 16.1% ..... 341 dpm/100 cm^2

Section 5.9, Page 5-95. The current text discusses the results of the RESRAD modeling with regard to development of both the subsurface and surface soil DCGLs but does not include a discussion of the results of the DandD modeling. The following discussion supplements that found in Section 5.9:

In order to derive the building surface DCGL, the DandD computer modeling code was iteratively run to arrive at the highest uniform concentration of residual surface radioactivity that results in an annual dose estimate to a single receptor in the critical exposure group that is equal to the regulatory limit (25 mrem/y)[2]. The following sections present the results of the computer modeling relating surface radioactivity source concentration with potential future dose.

The surface radioactivity DCGL is derived in consideration of the scenario in which a site worker is potentially exposed while on site performing leachate extraction and associated activities. Table A summarizes the results of modeling the projected future exposure potential for the scenario involving exposure while engaged in leachate extraction and associated activities at the Site. The isotope mixtures used are typical of, and consistent with, the measured isotopic mixture in soil at the site (Cabrera 2001). The Th-230 to Th-232 ratio used (3.1:1) is derived using a volume-weighted calculation that takes into account the range and volumetric significance of measured ratios found at the site.

A review of the computer modeling output of the DandD code (Attachment 3) reveals that exposure from the inhalation pathway dominates the potential future dose. The thorium isotopes (Th-232, Th-230, and Th-228) are the most significant contributors to

2 The DandD code lacks some of the flexibility and sophistication found in RESRAD. DandD reports peak annual doses based upon the upper 95% confidence interval for the 90th percentile dose calculated. While this metric is not precisely equivalent to the peak mean dose reported by RESRAD, it typically results in a more conservative correlation between dose and concentration of residual radioactivity.

total effective annual dose. Figure A and Figure B illustrate the relative pathway and isotopic contributions to total effective dose equivalent resulting from the building surface DCGL.

Table A Building Surface Radioactivity Source Term

Statistic	**Projected Annual Dose (mrem/year)**
Annual Dose Limit (10 CFR 20.1401, 1402)	25.0
90th Percentile	22.0
95% Confidence Interval about the 90th percentile	19.6 to 25.0
Computer printouts showing source term, dose, and radionuclide contribution distributions are in Attachment 3.	

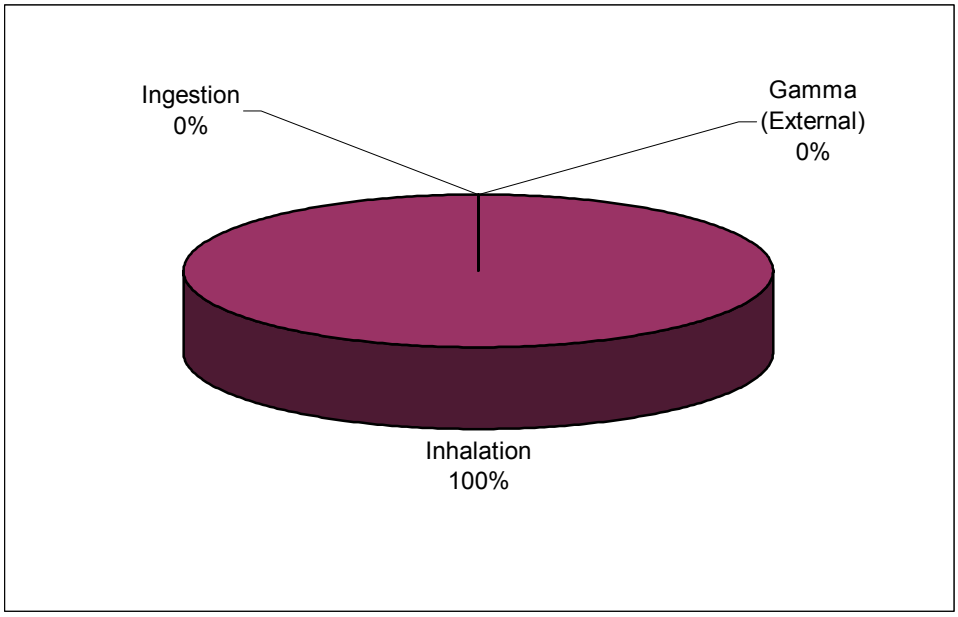

Figure A Pathway Contributions to Building Surface Source Term TEDE

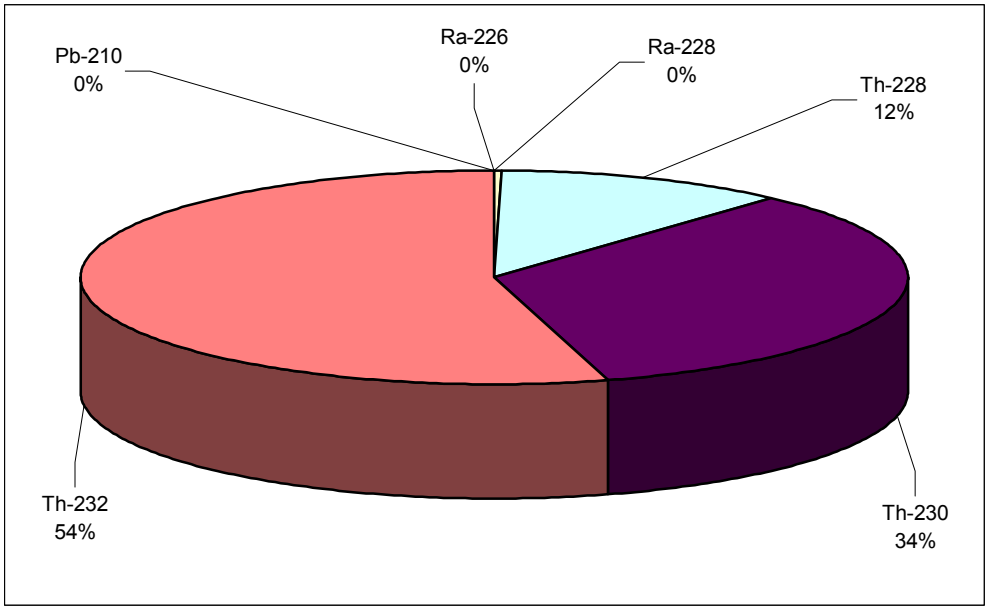

Figure B Isotopic Contributions to Building Surface Source Term TEDE

Section 8.0, Page 8-1, 1st ¶. The current text itemizes four objectives of the planned decommissioning activities. Removal of above grade LCTS piping, the LCTS building, and other miscellaneous structures was designed to serve the first two stated objectives. With the decision to leave these in place, objectives 1 and 2 should be replaced with a single objective to:

"quantitatively demonstrate that the concentration of residual radioactivity present on surfaces of the LCTS building, its structural appurtenances, and the former decontamination pad are below the surface radioactivity $DCGL_W$."

Section 8.0, Page 8-1, 3rd ¶. Per the request by MDEQ and the NRC's expressed desire to honor that request, MDNR no longer commits to terminating and abandoning the LCTS piping or to the removal of the LCTS building and decontamination pad. With the decision to leave these in place, the third paragraph should be replaced with the following paragraph:

A final radiological status survey of the LCTS building and the former decontamination pad is planned to demonstrate compliance with the decommissioning standard's annual dose limit for unrestricted use.

Section 8.1. Section 8.1 of the DP should be replaced in its entirety with the following set of paragraphs:

There are no contaminated structures located on site, and building/structure remediation activities are not planned. However, the LCTS building was used as a staging area and shelter during the performance of previous site characterization surveys and is currently used to temporarily store containerized, potentially contaminated personal protective equipment (PPE) and sample-derived waste. The location of the building is identified on the site map. Routine radiological surveys performed on site, including surveys of the building and its contents, have provided evidence that the building and its contents have not been radiologically contaminated by virtue of these uses.

Removal of the container of sample-derived waste and its associated radioactive materials is the first planned activity. Following removal of the containerized waste, MDNR will perform a final status radiological survey of the LCTS building's surfaces intended to satisfy the decommissioning dose criterion. It is not anticipated that radiological remediation tasks will be employed within the building because prior radiological surveys have shown the building to be radiologically clean, and routine radiological surveys of the container stored in the building continue to show that residual radioactivity associated with the sample-derived waste is contained. As a result, no special radiation protection methods and control procedures are planned for this work.

The concrete pad is also subject to the surface radioactivity DCGL. Like the LCTS building, it is not known or expected to be contaminated. MDNR will perform a final status radiological survey of the top surface of the decontamination pad to demonstrate compliance with the decommissioning dose criterion. The underside of the concrete pad is inaccessible for survey. However, it is known that the decontamination pad was built on the clean cover material after the radioactive slag had been isolated within the confines of the slurry wall and capped with several feet of engineered soil cover material. It is very unlikely that the underside of the decontamination pad could be impacted with measurable concentrations of residual radioactivity.

All of the activities described in this section will be performed by a contractor. The MDNR's radiation safety officer will retain responsibility for the oversight of radiological operations performed, and all licensed activities will be performed under the authority of the MDNR's radioactive materials license. MDNR commits to conducting decommissioning activities in accordance with written and approved procedures. There are no unique safety or remediation issues associated with the planned activities.

Section 8.2. Section 8.2 of the DP should be replaced in its entirety with the following set of paragraphs:

The only system or equipment present at the site is the LCTS. The LCTS system has never operated and licensed radioactivity has never been introduced to the system, therefore, there are no contaminated systems or equipment on site, and remediation activities are not planned. Portions of the installed LCTS piping (crocks, valve boxes, and extraction wellheads) do penetrate the cover such that they are visible above-grade.

Since the system has never been operated since it was installed many years ago, MDNR had planned to retire the LCTS piping as part of its planned decommissioning activities at the site with the goal of restoring the natural landscape and removing an attractive nuisance. However, both the MDEQ and S.C. Holdings expressed their desire for the system to remain in place.

Leaving the LCTS system in place does not pose a radiological issue for the MDNR since the LCTS system has never been operated and licensed radioactivity has never been introduced to the system. It does, however, introduce the possibility that the LCTS system could be made operational at some time in the future. MDNR has evaluated the radiological consequences of leachate extraction operations (including spills and leakage scenarios) in response to the NRC's request for additional information in connection with their review of this DP.

Section 8.5, Figure 8-2, Revised Schedule (Gantt Chart). A revised schedule is provided in Attachment 4.

Section 9.3.1, Page 9-4. Mr. Dennis Fedewa has replaced Ms. Kelli Sobel as the MDNR Agency Administrator of record for the Tobico Marsh SGA site NRC license.

Figure 9-2, Page 9-5, (Organization Chart). A revised organization chart is provided to reflect personnel changes in key management positions.

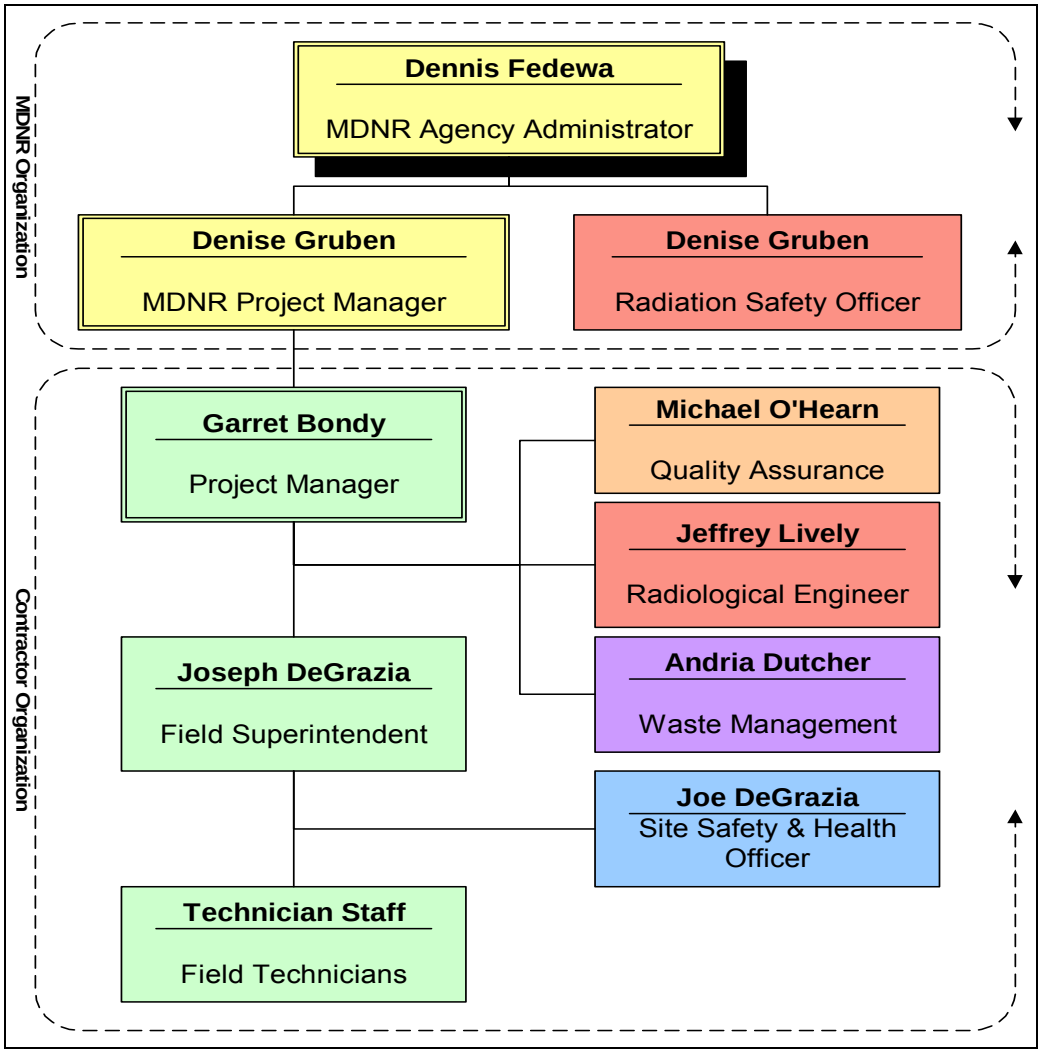

Figure 9-2 Decommissioning Management Positions

Section 9.3.4, Page 9-7. Mr. Garret Bondy has replaced Ms. Elena Goodhall as the Contractor Project Manager for the Tobico Marsh SGA site decommissioning project.

Section 9.3.6, Page 9-8. Mr. Joseph DeGrazia has replaced Ms. Elena Goodhall as the Site Safety & Health Officer for the Tobico Marsh SGA site decommissioning project.

Section 12.0, Page 12-1. Section 12.0 of the DP should be replaced in its entirety with the following paragraphs:

NUREG-1727 specifies that the licensee or responsible party have a program to manage radioactive waste generated as part of decommissioning. Under the "no action" alternative, the decommissioning activities yet to be performed at the site involve only sampling and survey activities and the removal of a container of sample-derived waste previously generated at the site. These activities are not expected to result in the generation of wastes containing licensable or measurable quantities of radioactive materials. As a result, there is no realistic potential for decommissioning activities to generate radioactive wastes.

The LCTS building currently stores one drum of sample-derived waste that may contain radioactive material. The drum will be shipped off site to be processed for appropriate disposal through a subcontracted waste broker licensed to possess and process radioactive materials. The following DP sections discuss characterization and disposal of the wastes, as appropriate.

Section 12.1.1, Page 12-1. Delete the second paragraph in Section 12.1.1.

Section 12.1.2, Page 12-2. Section 12.1.2 of the DP should be replaced in its entirety with the following paragraph:

The only solid waste generated during the decommissioning process is that associated with the single container of sample-derived waste that has been in storage in the LCTS building, and any sample-derived waste generated during the conduct of the final status radiological survey of the surface soils at the site. The existing container of sample-derived waste is assumed to be impacted with low levels of radioactivity. Sample-derived wastes generated in the course of the final status survey of the surface soils are not expected to be radioactive. The total volume of solid wastes generated during decommissioning is estimated as follows:

- Approximately 8 cubic feet of sample-derived waste (one 55-gallon drum)
- Approximately 1 cubic foot of unused soil sample fraction (60 soil samples, 500 ml each).

Section 12.1.4, Page 12-2. Section 12.1.4 of the DP should be replaced in its entirety with the following paragraphs:

The contents of the drum have not been characterized for waste disposal purposes, but the outside of the drum has been subjected to routine radiological surveys to ensure no radioactive contamination has escaped. The drum contents will be characterized using

existing knowledge and data, and based on that characterization, the drum will be assessed to ensure it meets the transportation requirements of the U.S. Department of Transportation (DOT). If the drum does not meet the applicable DOT requirements, it will be overpacked or the contents will be repackaged, as appropriate.

Items and debris confirmed to meet the applicable decontamination limits for radiological contamination will be disposed of as sanitary waste in a nearby landfill. Items and debris determined to be radiologically contaminated, or which cannot be positively identified as meeting the applicable decontamination limits for radiological contamination, will be packaged in DOT-compliant containers and will be stored on pallets, under tarps, and within the fenced area until disposal can be arranged. Such materials will be disposed of at a licensed, low-level radioactive waste disposal facility. Transportation and disposal of any radiological waste will be accomplished through a waste broker, using the broker's existing disposal contract (likely with Envirocare of Utah).

Section 14.1, Page 14-1. Section 14.1 of the DP should be replaced in its entirety with the following paragraphs:

Radiological release criteria for the Tobico Marsh SGA site are derived from appropriate dose modeling as described in Section 5.0 of the DP and are based upon the decommissioning dose limit for unrestricted use of the site following license termination (NRC 1997a). The final status site release criteria are applicable to radiologically impacted structures, components, and soils that are to be left in place as a feature of the final condition of the site at the time of request for license termination. The projected final site condition at the completion of decommissioning activities is essentially as it currently stands; the building and decontamination pad will remain in place and the disposal cell will not be altered in any significant way.

There are three distinct radiological DCGLs for the site. One DCGL is derived for the surfaces of the LCTS building and the decontamination pad. Two others are provided for the potentially impacted soils at the site (one for surface soil, the other for subsurface soil).

Section 14.1.1, Page 14-1. Section 14.1.1 of the DP should be replaced in its entirety with the following paragraphs:

14.1.1 Final Status Survey Release Criteria for Structural Surfaces

To ensure that the standard is not exceeded, dose modeling was preformed to arrive at the maximum uniform surface activity concentration ($DCGL_W$) corresponding to the 25 millirem annual dose limit. The structural surfaces $DCGL_W$ applies to the surfaces of the LCTS Building, it structural appurtenances, and the concrete decontamination pad.

The radionuclide composition of source term used in the computer modeling code includes those isotopes in the Th-230 and Th-232 decay series that have relatively longer half-lives (greater than 180 days). This does not mean that potential radiation dose from shorter-lived progeny in the decay series that were not directly included in the source term is not accounted for. As a matter of course, the computer code appropriately assumes that the shorter-lived progeny are present in equilibrium with their parent nuclides and accounts for the dose they contribute using dose conversion factors that include dose from both the parent and progeny. This means that the total activity input to the model does not equal the total activity of the defined source term.

Given that the final status survey will rely on measurements of the amount of radioactivity present on the building's and other surfaces without regard to it's isotopic speciation, it is necessary to relate the source term used in the model to the total alpha and beta radioactivity to arrive at an appropriate surface activity DCGL. The total surface radioactivity corresponding to the modeled source term in transient equilibrium is 4641 dpm/100 cm^2. Since survey instruments used to measure surface radioactivity are designed to respond independently to either alpha or beta surface radioactivity, it is also necessary to segregate those isotopes that decay by alpha emission from those that decay by beta particle emission.

Table B shows the contribution of each radionuclide assumed present in the source term to the total surface radioactivity and further segregates those isotopes that contribute to the beta radiation signal from those contributing to the alpha radiation signal. From Table B it can be discerned that the applicable surface activity $DCGL_W$ for the building and other structures at the site is:

- 3,209 dpm/100 cm^2, total alpha radioactivity, or
- 1,432 dpm/100 cm^2, total beta radioactivity

Table B

Th-230 Series			
Nuclide	dpm/100 cm^2	Alpha Activity	Beta Activity
Th-230	1060	1060	
Ra-226	23	23	
Rn-222	23	23	
Po-218	23	23	
Pb-214	23		23
Bi-214	23		23
Po-214	23	23	
Pb-210	11		11
Bi-210	11		11
Po-210	11	11	
Total	1231	1163	68
Th-232 Series			
Nuclide	dpm/100 cm^2	Alpha Activity	Beta Activity
Th-232	341	341	
Ra-228	341		341
Ac-228	341		341
Th-228	341	341	
Ra-224	341	341	
Rn-220	341	341	
Po-216	341	341	
Pb-212	341		341
Bi-212	341	123	218
Po-212	218	218	
Tl-208	123		123
Total	3410	2046	1364
TOTAL ACTIVITY Building Surface DCGL		Gross Alpha (dpm/100 cm^2)	Gross Beta (dpm/100 cm^2)
		3209	**1432**

Section 14.1.3, Table 14-3, Page 14-3. Table 14-3 of the DP should be revised to include a $DCGL_{EMC}$ for the LCTS Building and other surfaces final status survey. Since each of the four survey units with a potential for surface deposited radioactivity are classified as "Class 3" survey units, the $DCGL_{EMC}$ is set at the $DCGL_W$. Replace Table 14-3 in its entirety with the following Table:

Table 14-3 Media Specific $DCGL_{EMCs}$

Media	Radionuclide	$DCGL_{EMC}$
Surface Soil	Th-232	357 pCi/g
Building (Structural) Surfaces	Total Alpha	3209 dpm/100 cm^2
	Total Beta	1432 dpm/100 cm^2

Section 14.4.1, Page 14-20. Section 14.4.1 of the DP should be replaced in its entirety with the following paragraphs:

LCTS Building Survey

The LCTS building and its appurtenant surfaces will be surveyed for the presence of residual surface radioactivity associated with the thoriated slag source term at the site. The survey will be performed after the 55-gallon drum stored in the posted RMA within the LCTS building has been removed. The survey will be performed with standard portable radiation monitoring equipment capable of measuring the beta (or alpha) emissions associated with thorium and its progeny. The survey will consist of direct static measurements of the surfaces at randomly selected locations.

Decontamination Pad Survey

The decontamination pad is a concrete slab poured on the finished cell cover after the deposited radioactive material had been encapsulated within the cell. Consequently, the decontamination pad is not potentially impacted by radiological operations from historical activities during the placement of the thoriated slag at the site. The decontamination pad was used briefly by MDEQ, but routine radiological surveys performed since then have not shown residual radioactivity to be present.

The surface of the decontamination pad will be surveyed for the presence of residual surface radioactivity associated with the thoriated slag source term at the site. The survey will be performed with standard portable radiation monitoring equipment capable of measuring the beta emissions associated with thorium and its progeny. The survey will consist of direct static measurements of the surface arrayed on a systematic grid with a random starting point overlying the slab.

Measurement Methods

Structure surfaces (including the decontamination pad) will be measured in the field using standard portable radiation measurement equipment. MDNR plans to use the Eberline E600 multipurpose portable radiation survey instrument coupled with a SHP-360 probe to perform the surface activity surveys. Where discrete measurements are specified, the instrument will be operated in the scaler mode for a fixed measurement

time interval. Scans, where performed, are made with the instrument in the "ratemeter" mode.

Based on dose modeling performed using the NRC's DandD code, surface deposited gross alpha and gross beta surface activity DCGLs have been derived. Using the isotopic mixture ratios described in the DP, the DandD code was used to derive the building surface concentration corresponding to a peak annual dose of 25 mrem. Site-specific data obtained during the extensive subsurface soils characterization work form the basis for the ratio of thorium 230:232 used in the dose model. The thorium–230 and thorium–232 decay series were decayed for fifty years to ascertain the appropriate values of longer-lived progeny included in the modeled source term. Short-lived progeny are included in the dose conversion factors used in the model to calculate dose.

The two thorium decay series were then analyzed to differentiate the total alpha emission rate from the total beta emission rate for the source term concentration corresponding to 25 mrem per year (See Table B). As indicated in Table B, the Th-230 series total activity input to the DandD model was 1231 dpm/100 cm^2 and the Th-232 series total activity input to the DandD model was 3410 dpm/100 cm^2, for a combined total activity of 4641 dpm/100 cm^2. For every 4641 disintegrations, 3209 result in alpha emission, while the remainder, 1432, results in beta emission. Therefore, the total alpha surface activity $DCGL_W$ corresponding to an annual effective dose equivalent of 25 mrem is 3209 dpm/100 cm^2. The corresponding total beta surface activity DCGL is 1432 dpm/100 cm^2.

Sample Size Determination

MDNR has planned to make a sufficient number of measurements such that the median residual surface radioactivity on the building and other structural surfaces can be shown to be less than (or equal to) the surface activity DCGL with 95% confidence.

In the case of evaluating compliance with the $DCGL_W$, a sample size must be adequate to determining whether the mean (average) survey unit surface concentration is less than the $DCGL_W$, given the pre-defined acceptable decision errors. Of course, the mean concentration could be either greater than or less than the $DCGL_W$. An insufficient sample size would provide inadequate power to discern that the true mean concentration in the survey unit was less than the $DCGL_W$ even when that condition was true.

The estimated minimum sample size required to determine whether the mean residual surface radioactivity concentration in a survey unit is independent of survey unit size and can be calculated once and applied to each survey unit. The minimum sample size required to estimate the mean concentration in a survey unit is computed using standard formulas for the one-sample Sign test (MARSSIM) and is presented below. The sample size calculation is designed to be used with the gross beta activity $DCGL_W$ and assumes:

- A standard deviation (σ) of 250 dpm/100 cm^2 for the sample set used to determine the mean (median) total surface residual radioactivity concentration (i.e., a coefficient of variation[3] of 1 if the true mean is 250 dpm/100 cm^2).

- A shift (Δ) of 300 dpm/100 cm^2 is determined to be significant for the mean (median) total surface residual radioactivity concentration. The shift is the width of the gray area below and above which uncertainties in discrimination are critical to the decision maker. The gray area is bounded by the "lower boundary of the gray region" (LBGR) and the $DCGL_W$. The shift defines the decision maker's critical window of observation and is based on the decision maker's acceptance of consequences of making Type I and Type II errors in testing the null hypothesis.

- The relative shift (Δ/σ) is the ratio of the shift and standard deviation. The calculated value of relative shift for the mean (median) total surface residual radioactivity concentration is 1.0 (i.e., 300 dpm/100 cm^2 / 250 dpm/100 cm^2 = 1.2).

- Null hypotheses (H_o) of:

  - *The mean (median) total beta residual surface radioactivity concentration is greater than 1432 dpm/100 cm^2*

  This is the conservative form of the null hypothesis in that it places the burden of proof on MDNR to demonstrate that the average residual radioactivity concentration in the survey unit is less than the $DCGL_W$ (MARSSIM).

- False negative err rate = 0.05 (i.e., alpha = 0.05). This ensures that there will be no greater than a 5 percent chance of incorrectly rejecting the null hypothesis and finding that a survey unit mean (median) surface residual radioactivity concentration is *less than* the $DCGL_W$ when, in fact, it is greater than the $DCGL_W$.

- False positive err rate = 0.05 (i.e., beta = 0.05). This ensures that there will be no greater than a 5 percent chance of incorrectly accepting the null hypothesis and finding that a survey unit mean (median) surface residual radioactivity concentration *exceeds* the $DCGL_W$ when, in fact, it is less than the $DCGL_W$.

Computed minimum sample size per survey unit is calculated assuming the sampling statistics itemized above and using the sample size calculations described in MARSSIM (NRC 2000). The minimum sample size is tabulated in Table C. The computations are shown in the following equations:

$$\Delta / \sigma = \frac{(\text{DCGL - LBGR})}{\sigma_s} = \frac{(1{,}432 - 1{,}132)}{250} = 1.2 \qquad (1)$$

3 Coefficient of Variation = standard deviation/mean

The "Sign p" value is an intermediate statistic used to determine the minimum sample size. The Sign p is the estimated probability that a random measurement from the survey unit will be less than the DCGL when the survey unit median is actually at the Lower Boundary of the Gray Region (LBGR) value selected. The Sign p value for a relative shift of 1.2 is picked from MARSSIM, Table 5.4, *Values of Sign p for Given Values of Relative Shift, Δ/σ, when the Contaminant is Not Present in Background*. The Sign p for a relative shift of 1.2 is 0.884930.

The Z statistic is a percentile score corresponding to the accepted probability of decision error at the DCGL and LBGR. The specified acceptable probability of decision has been selected as 0.05 for both α and β. Consequently the Z statistic for $Z_{1-\alpha}$ and $Z_{1-\beta}$ are the same value, 1.645.

The number of data points, N, to be obtained to satisfy the Sign test with sufficient statistical power is calculated using the following formula:

$$N = \frac{(Z_{1-\alpha} + Z_{1-\beta})^2}{4(Sign\ \mathrm{p} - 0.5)^2} \tag{2}$$

$$N = \frac{(1.645 + 1.645)^2}{4(0.884930 - 0.5)^2} = \frac{10.8}{0.59} = 18.3 \tag{3}$$

To account and compensate for uncertainty in the computations of minimum sample size, as well as the possibility that some sample data may be lost or deemed unusable due to analytical and sampling error, anomalous results which are judged to be erroneous, and other errors, minimum sample size computations are increased by 20 percent and rounded up to obtain sufficient data points to yield the desired power.

Table C Computed Minimum Sample Size per Survey Unit (Sign Test)

Derived Concentration Guideline Level	Computed Minimum Sample Size	Sample Size With 20% Margin
Mean Total Beta Surface Residual Radioactivity Concentration	19	23
alpha = 0.05, beta = 0.05, relative shift = 1.2		

As expected, non-parametric tests require a greater sample size than a conventional normal means test, but liberates the decision maker from the need to meet the underlying assumption basis of normality. Indeed, the sample data are not expected to be normally distributed. Consequently, the sample plan design assumes the need for non-parametric assessment of the data.

Sample Distribution

The sampling design software *Visual Sample Plan* (VSP), Version 2.5 (PNNL 2005), will be used to randomly lay out the required number of samples or measurements over the

survey units. Simplified drawings of the disposal cell cover, the LCTS building, and the decontamination pad will be prepared and used with VSP to produce the actual measurement layout for each survey unit or designated stratum.

A summary of the survey units identified for the FSS at the site along with the planned number of samples or measurements needed to assess the mean residual radioactivity concentration in each is provided in Table D.

Table D Planned Sample Sizes by Survey Unit

Survey Unit ID#	Description / Location	Planned Number of Samples / Measurements	Survey Unit Classification
TOBICO–02	LCTS Building Interior Surfaces	23	3
TOBICO–03	LCTS Building Exterior Surfaces	23	3
TOBICO–04	LCTS Building Appurtenances (Concrete pad and walkway)	23	3
TOBICO–05	Decontamination Pad Surface	23	3

Data Analysis Framework

The data analysis framework is critical to sample plan development because it establishes the basis for decision and drives the sample size. The evaluation process will use an analysis structure incorporating three possible common statistical procedures as well as conventional qualitative and semi-quantitative comparisons. The tests are:

- **Sign Test—**The Sign Test is a one-sample, non-parametric test that can be used to evaluate compliance with the $DCGL_W$. The Sign Test is the recommended compliance evaluation procedure when the contaminant(s) under evaluation are not present at significant levels in background. While the thorium series radionuclides clearly exist in nature, because background concentrations are appreciably lower than the DCGL(s), MDNR does not feel that it is necessary to establish a reference area and distinguish potential contaminant concentrations from background[4]. Additionally, any combination of the individual samples (each individual survey unit is a "sample" in this context) can be compared to the DCGL with the Sign Test. For example, the data from the LCTS Building interior and exterior survey units could be pooled together for an overall building comparison to the DCGLs.

4 It is always possible that the influence of naturally occurring radioactivity from earth materials may affect the decision process. If, at a future time, it is felt that establishing a reference area to evaluate the significance of local background effects would benefit the decision process, this can be accomplished at that future time in accordance with MARSSIM (NRC 2000).

- **Normal Means Test—**This is the traditional two-sample t-test based on the central limit theorem (i.e., normality). It can be used to assess compliance, derive confidence intervals, and compare between samples (e.g., mean total surface radioactivity concentration in one survey unit vs. the same parameter measured in another survey unit) when both sample distributions are normal or *do not* deviate appreciably from normality. Provided the data sets under evaluation are normally distributed, the normal means test has the advantage of providing significantly more power than the comparable non-parametric tests, given the same number of samples.

MDNR expects to use the Sign Test to evaluate compliance with $DCGL_W$s. In addition to these inferential tests, data analysis will include qualitative visual analysis (e.g., histograms, scatter diagrams, box and whisker plots). Additional analytical methods (e.g., spatial correlation) as well as spatial analysis (e.g., posting on diagrams, iso-concentration plots) not required to support the decision rule are not explicitly planned for but could be employed on an ad-hoc basis to gain insight.

The data analysis framework will incorporate data quality analysis (DQA) components discussed in MARSSIM (NRC 2000) and EPA guidance (EPA 1992) to assess the overall usability of the data for its intended purpose. The data evaluation process will be validated, and statistical analysis methods will be used, to assess whether variability and bias in the data are small enough to allow MDNR to use the data to support the sampling objective—release of the MDNR site from radiological control through license termination. Risk managers will be presented with an ensemble of information, logically interpreted, and supported by rationale to gauge compliance.

Sample Allocation Protocol

As previously described, sample locations for the FSS have been randomly selected and designed to meet the specific objective of the data collection activity. Maps detailing the selected sample locations for each survey unit will be provided at the time of the FSS in the field work plan. The objective of the FSS is the unbiased assessment of the residual radioactivity present in various survey units of the site. It is statistically inappropriate to bias the sample or measurement locations selected for the survey to achieve this objective.

Static surface measurement locations for survey units Tobico-02 through -04 will be randomly chosen using VSP. The sample allocation technique used to distribute the required number of measurements for the decontamination pad (survey unit Tobico-05) is slightly different. While considered a class 3 survey unit, a systematic triangular grid with a random starting point will be used to select the measurement locations. The systematic grid was selected for this survey unit because there were no prior, documented, radiological surveys of the concrete decontamination pad. The grid system ensures uniform spatial coverage, bounds the size of locally elevated surface radioactivity anomalies, and allows for post survey evaluation of any surface radioactivity anomalies that might be encountered.

Measurement Sensitivity

Measurement sensitivity is an important component of the sampling and analysis plan because it is critical that measurement systems be capable of detecting the benchmarks that guide decisions including the DCGL comparisons. This section discusses measurement system sensitivity in light of the specific benchmark comparisons.[5]

Field Instrument for Direct Static Measurement of Building Surfaces

The direct measurement field instrumentation specified is a reliable device with adequate detection sensitivity and is suitable for timed static field measurements to compare with the total surface contamination concentration DCGL. The following formulation is used to predict the minimum detectable concentration (MDC), in dpm/100 cm^2, for the E-600 survey instrument using the Eberline SHP-360 Geiger-Mueller "pancake" detector probe.

$$MDC = \frac{3 + 4.65\sqrt{C_b}}{T_s \times \frac{A_P}{100\,cm^2} \times \varepsilon_s \times \varepsilon_i} \tag{4}$$

Where: MDC = the minimum surface activity concentration above background radioactivity (in dpm/100 cm^2) that can be measured with 95 percent confidence.

C_b = the total number of background counts over the sample count period (T).

T_s = Sample count time (in minutes).

A_P = Probe size (in cm^2).

ε_s = Surface Efficiency

ε_i = Counting system efficiency (counts/disintegration).

Using conservative estimates of the parameters affecting the MDC of the static field measurement, an *a priori* assessment of the MDC can be determined. This value represents the worst plausible case measurement conditions and yields the highest expected measure of the detection sensitivity for the analysis. As such, the *a priori* estimate of the MDC serves as a figure of merit about the capability of the measurement. Table E and the following calculations define the *a priori* MDC estimates for the static beta surface radioactivity measurements using the E-600 and the SHP-360 detector probe identified.

5 Measurement sensitivity computations are derived from the basic detection limit relationship LD = k + 4.65B. This relationship as derived by Curie (1968) set the constant k at 2.71. Since that time it has been shown (Brodsky 1992) and generally accepted that a constant factor of 3 is more appropriate. This plan calculates field measurement sensitivity using the constant factor 3.

Table E Static Surface Radioactivity Measurement

Parameter		Value Used	Remarks
C_b	Background Counts	40	Value used is the product of the maximum expected background count rate (40 cpm) and sample count time (one minute).
T_s	Sample count time (in minutes)	1.0	Count time programmed into the calibrated instrument specifically for this sampling event
A_P	Probe size	15	In cm^2.
ε_s	Surface efficiency	1.0 (100%)	Since the Instrument efficiency is determined as the 4π efficiency, surface efficiency is used only to account for the effects of backscatter and surface attenuation of the beta signal. For concrete surfaces, the contribution from backscatter and degradation from surface attenuation are known to roughly cancel one another out when the beta energy approximates that of Cl-36. In addition, the use of an aluminum backed, anodized surface calibration source (as opposed to an electroplated source on a stainless steel backing closely approximates the surface efficiency characteristics of concrete.
ε_i	Instrument system efficiency in counts/disintegration	0.25 (25%)	Nominal 4π beta efficiency for the SHP-360 thin window GM probe determined with a Cl-36 calibration source is 25%. Actual efficiency for each individual probe is programmed into the memory chip of the probe's smart pack.

These values predict a worst plausible case MDC for the static field measurement to be 864 dpm/100 cm^2 total beta activity as shown in the following calculation.

$$MDC = \frac{3 + 4.65\sqrt{40}}{1.0 \times 0.15 \times 1.0 \times\ 0.25} = 864 \text{ dpm/100 cm}^2 \qquad (5)$$

The *a priori* total beta radioactivity MDC over the range of expected conditions is lower than the total beta residual surface radioactivity concentration DCGL of 1,432 dpm/100 cm^2 by a factor of approximately 1.6). Appropriate sensitivity is a key requirement of quality data measurement methods. The *a priori* "worst-case" MDC indicates that the objective to use measurement methods and instruments with MDCs (a measure of sensitivity) well below the DCGL benchmark being measured is achievable (NRC 2000).

In practice, the instrument used for field measurement will be calibrated to respond directly in units of dpm/100 cm^2. As such, background collected in the field will be presented in these units instead of counts or cpm. Nominally, a background count rate of 30 beta cpm yields an instrument background of approximately 800 dpm/100 cm^2. The fact that the instrument presents the background activity in units other than counts or cpm does not change the counting statistics of the measurement and does not affect the MDC of the instrument. Background measurements in the field will be made using the scaler mode algorithm built into the E-600 instrument.

The foregoing calculation demonstrates that surface activity instrument has an MDC adequate to detect residual radioactivity concentrations well below the $DCGL_W$.

Furthermore, if necessary, sensitivity can be augmented readily in the field by increasing the count time. Since sensitivity and MDC are related to and significantly influenced by background, establishing instrument background on a frequent periodic basis during measurement activities is prudent.

Quality Control Data for Field Survey Measurements

The Final Status Survey for building and structural surfaces relies on *in situ* field measurements using conventional health physics measurements and practices. All data necessary to support the DQO decision requirements for building and structural surfaces will be provided by the measurement techniques discussed. MDNR does not plan to collect any media samples form the LCTS building or the decontamination pad (e.g., concrete from floors or ceiling materials) for laboratory analysis.

The most appropriate QC method to assess the potential error that might occur with direct radiological measurement of a surface is the replicate field measurement. For *in situ* measurements, replicate measurements will be obtained by performing a second measurement at the same measurement location using the same instrument to measure method precision. In practice, the technician will simply leave the detector in the randomly selected location and log the data for a second count time. Replicates are specified in accordance with the 1 in 20 rule commonly applied in the environmental industry and cited in guidance and MARSSIM (EPA 1988, NRC 2000). Table F illustrates the schedule of planned replicate measurements, based on the scheduled number of direct static measurements.

Inspection of Table F reveals that MDNR intends to take more replicate measurements (12 versus 5 per the 1:20 rule) than might typically be necessary due to the limited nature of the building surfaces and structure surfaces being assayed during this FSS, and MDNR's desire to firmly establish the quality of the FSS data set. Replicates will be allocated throughout the survey units being measured, as necessary, according to the scheduled number described in Table F.

Table F Numbers of Direct Static Surface Measurements & Associated Replicate Measurements

Survey Unit	Number of Measurements Scheduled	Number of Replicates Scheduled
TOBICO–02 LCTS Building Interior Surfaces	23	3
TOBICO–03 LCTS Building Exterior Surfaces	23	3
TOBICO–04 LCTS Building Appurtenances (Concrete pad and walkway)	23	3
TOBICO–05 Decontamination Pad Surface	23	3
Total	92	12