Rio Grande Resources, Corp.
Mt Taylor Mine
12/27/21

Addendum to
"Work Plan for Post-Mining Radiological Surveys of Permit Area and Impacted Lands"
Mount Taylor Mine
Cibola County, New Mexico

Rio Grande Resources Corp. (RGR) submitted the "Work Plan for Post-Mining Radiological Surveys of Permit Area and Impacted Lands" (Work Plan) on June 8, 2020 to New Mexico Mining and Minerals Division (MMD). The Work Plan was a requirement of Section 9.L.3 of the Mt Taylor Mine Permit CI002RE, Rev 13-2.

RGR received a comment letter from MMD and New Mexico Environment Department (NMED) dated October 28, 2021, which presented comments regarding the Work Plan. RGR prepared responses to those comments. To more adequately address those comments, RGR has prepared this addendum, which contains details and clarifications of its responses presented as proposed changes and additions to the Work Plan.

The following proposed revised sections of the Work Plan include:

1) Acronyms and Abbreviations (response to Comment #1)
2) Proposed Changes to Section 2 of the Work Plan (response to Comment #2)
3) Enlarged Figure 3 (response to Comment #6)
4) Data and Results of the Regression Analysis, 2012 Study (response to Comment #10)
5) Enlarged Figure 7 (response to Comment #11)
6) Proposed Changes to Section 4 of the Work Plan (response to Comment #13)
7) Proposed Changes to Section 6.1 of the Work Plan: Sign Test Discussion (response to Comment #18)
8) Appendix A - Derivation of Release Criteria for Surface Contamination (response to Comments #2)

1) Page of acronyms and abbreviations for the Work Plan (response to Comment #1):

ACRONYMS AND ABBREVIATIONS

µR/hr	Micro-Roentgen per Hour (exposure rate)
AF	Area Factor
Chevron	Chevron Resources Company
DCGL	Derived Concentration Guideline Level
FSS	Final Status Survey
GPS	Global Positioning Systems
Gulf	Gulf Mineral Resources Company
HPIC	High-Pressure Ionization Chamber
HSA	Historical Site Assessment
km	Kilometer
m	Meters
MARSSIM	Multi-Agency Radiation Survey and Site Investigation Manual
MDC	Minimum Detectable Concentration
MMD	Mining and Minerals Division, New Mexico Energy Minerals and Natural Resources Department
MWTU	Mine Water Treatment Unit
NaI	Sodium Iodide
NMED	New Mexico Environment Department, Mining Environmental and Compliance Section
NRC	U.S. Nuclear Regulatory Commission
pCi/g	Picocuries per Gram (activity concentration for solid matrices)
PMLU	Post-Mine Land Use
PRRL	Post-Reclamation Radiation Level
Ra-226	Radium-226
RCB	Radiation Control Bureau, NMED
RGR	Rio Grande Resources
RPP	Radiation Protection Program
RSS	Remedial Support Survey
s	Second
SOP	Standard Operating Procedure
UTV	Utility Terrain Vehicle
VSP	Visual Sample Plan

2) Proposed Changes for Section 2 of the Work Plan (response to Comment #2):

A) Label the original text in section 2 as “2.1 Criteria for Release of Land Areas”

B) Create sub-section “2.2 Criteria for Release of Buildings and Infrastructure”, with the following text:

2.2 Criteria for Release of Buildings and Infrastructure

As referenced in Condition 9.L.2 of Mine Permit CI002RE (Permit Revision 13-2, 2017), the dose-based license termination criterion specified in 20.3.4.426 NMAC (≤ 25 mrem/yr) applies to all mine facilities to be retained for the PMLU. While the PMLU for land areas is livestock grazing, the anticipated post-mine use of buildings expected to remain after Site closure is industrial or commercial (RGR, 2013). Because structural materials and equipment that meet the criteria for release could potentially be repurposed for domestic onsite uses or removed from the Site for use by third parties and/or to be sold for scrap, the release criteria described in the following Sections were conservatively developed for a potential residential exposure scenario as detailed Appendix A.

2.2.1 SURFACE CONTAMINATION RELEASE CRITERIA

Based on the 25 mrem/yr dosimetric release criterion specified in Condition 9.L.2 of Mine Permit CI002RE [20.3.4.426.(B) NMAC], RGR has developed a derived concentration guideline level (DCGL) for surface contamination from uranium and its decay products (in equilibrium) on the surfaces of buildings, including structural components (e.g. walls, floors, etc.), operational infrastructure (e.g. plumbing, pipes, HVAC systems, electrical power infrastructure, etc.), and operational equipment (e.g. water treatment systems and equipment, and furnishings/equipment associated with offices, maintenance shops, laboratory areas, storage areas, etc.).

The overall radiological criterion for release of buildings at the Mount Taylor Mine for unrestricted use, based on the conservative exposure scenario described in Appendix A, is equivalent to an average gross alpha radiation surface activity DCGL of 2,364 alpha decays per minute (dpm) per 100 cm^2.

Given that alpha-emitting and beta-emitting uranium progeny at the mine are in equilibrium, RGR will not be monitoring specifically for beta-emitters. This practice is allowed by NRC at uranium recovery facilities in those areas were alpha and beta-emitters are in equilibrium. For Mt. Taylor Mine, this method is appropriate due to several other factors including:

1) Detectors commonly used to measure beta radiation are sensitive to both gamma and beta radiation. The presence of gamma radiation can contribute to the beta count rate by ionizing atoms in the nearby materials (e.g. metal instrument housings), resulting in scattered electrons that can inaccurately indicate higher beta counts from the surface being measured.

2) Alpha radiation detectors produce lower, less variable background results than do beta detectors. They also have higher intrinsic detection efficiencies than do beta detectors. Hence, the alpha detectors provide more reliable results with lower minimum detectable activities (MDA).

Because the DCGL calculation assumes secular equilibrium for both alpha and beta emitters in the Uranium decay series, if the alpha criterion is met, the beta criterion will theoretically be met as well. In addition, the primary dose pathway for surface contamination is inhalation of particulate alpha radiation, which is twenty times more significant than beta radiation in terms of linear energy transfer and potential absorbed dose. Thus, relying on alphas for determining compliance with the 25 mrem/yr dose standard is conservative and appropriate.

For the above reasons, beta radiation measurements will not be quantitatively compared to a release criterion, but will be qualitatively evaluated as part of general survey objectives, for example to help identify areas of potential contamination where accumulated dust, paint, or other thin films of material may shield surface alpha emissions from detection with survey instruments. Measurements of gamma radiation will be evaluated in a similarly qualitative manner, for example to identify potential shielded deposits of radioactive material inside of inaccessible equipment or infrastructure such as pipes, drains, pumps, etc.

With respect to *removable* surface contamination, the modeling assumed 10% of the surface activity is removable and available for ingestion and inhalation pathways. Limited, yet representative, swipe testing results will be used to evaluate this modeling assumption. A detailed description of the derivation of radiological release criteria used is provided in Appendix A.

Finally, surfaces and equipment exceeding the DCGL will be removed from the facility and segregated for eventual placement in the South Waste Rock Disposal Cell.

2.2.2 VOLUMETRIC CONTAMINATION RELEASE CRITERIA

For distributed, volumetric contamination in building materials or associated equipment (e.g. concrete, pipes, valves, tanks, etc.), NUREG-1757 Vol. 2 indicates that the dose-based criterion for license termination (25 mrem/yr) also applies. Based on an American National Standard from the American National Standards Institute/Health Physics Society (ANSI/HPS, 2013), a volumetric concentration of 3 pCi/g of natural uranium (U-nat) and associated decay progeny in equilibrium equates to a de-minimis dose of 1 mrem/yr. This criterion, which is based on conservative assumptions regarding potential exposure scenarios for materials/equipment, is equivalent to a dose conversion factor of 0.33 mrem/yr per pCi/g. Scaling this conversion factor against the license termination criterion specified in 20.3.4.426 NMAC (≤ 25 mrem/yr) gives a volumetric release criterion of 75 pCi/g of U-nat in equilibrium with its decay progeny.

However, this criterion is applicable only for exposure scenarios consistent with those used in developing the ANSI/HPS standard, which involved only 500 hours of exposure per year. Scaling the calculated volumetric release criterion of 75 pCi/g against 8760 hours of exposure per year (100% occupancy), the equivalent release criterion would be 4.3 pCi/g, which is close to the 5 pCi/g cleanup standard for Ra-226 in surface soils as specified in MMD guidance applicable to the mine permit (MMD/NMED, 2016). Even for the most conservative exposure scenario, it is unrealistic to assume exposure to volumetrically contaminated materials 100% of the time and thus, for the purposes of the Work Plan, the volumetric release criterion will be set at 5 pCi/g for Ra-226 in equilibrium with the U-238 decay series.

3) Figure 3 has been enlarged for readability (response to Comment #6):

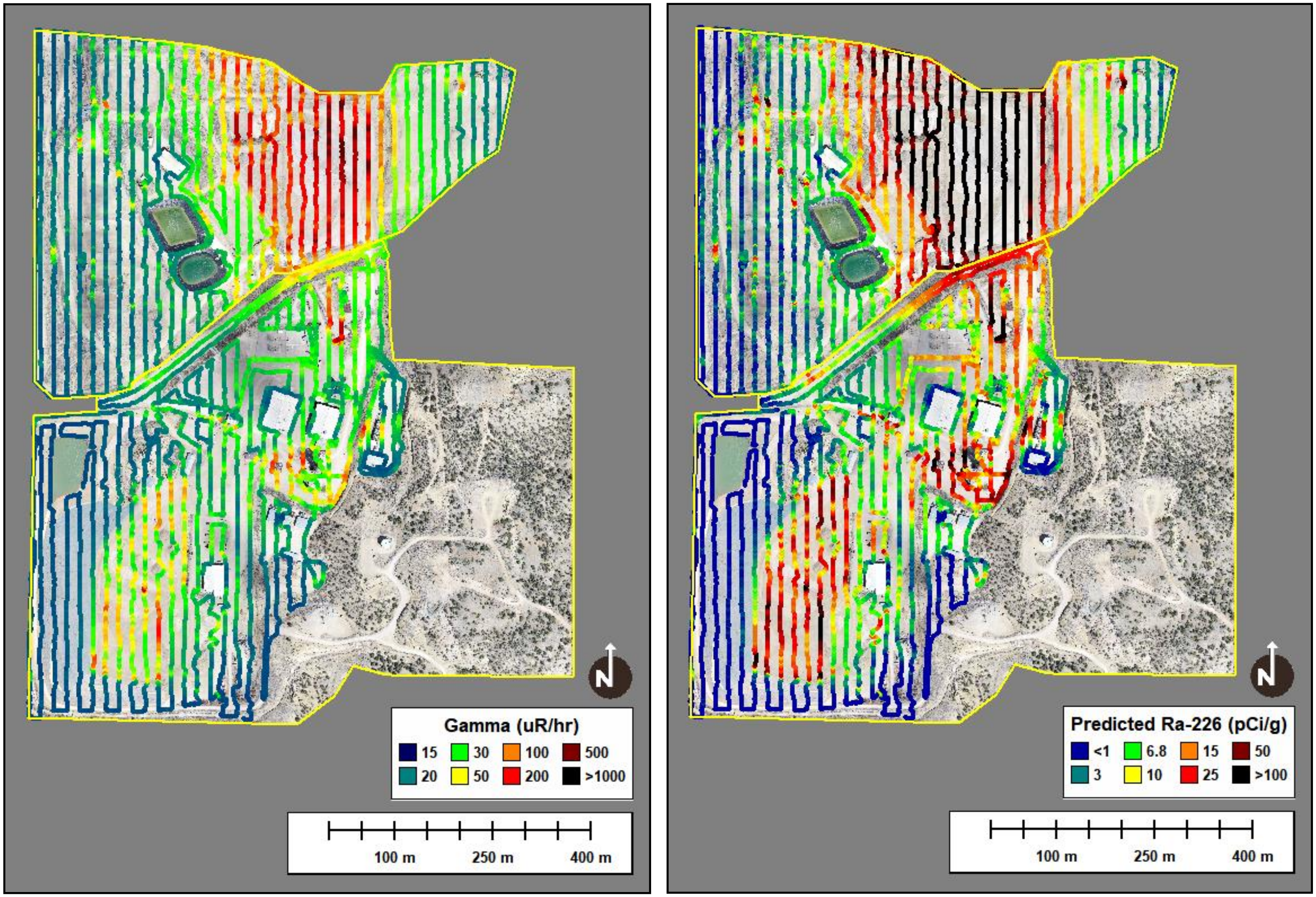


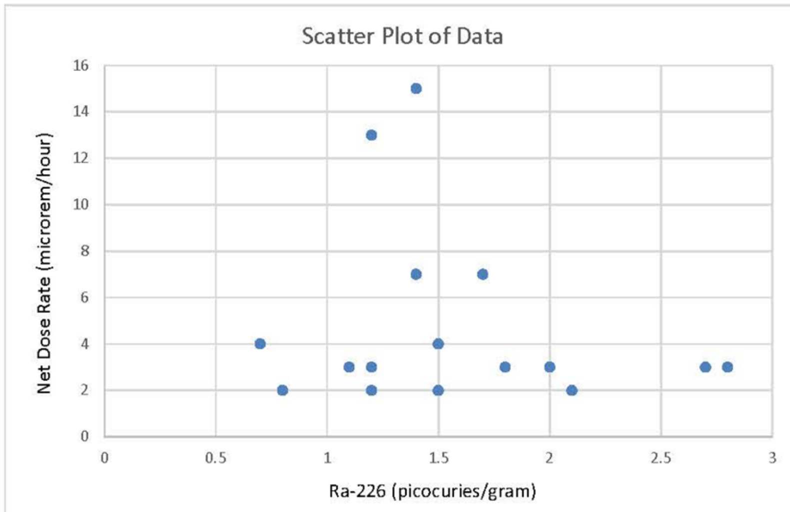
Figure 3: Measured gamma exposure rates (left) and respectively predicted Ra-226 concentrations in surface soil (0-15 cm) (right) across operationally Controlled Areas at the Mt. Taylor Mine.

4) Data and Results of the Regression Analysis of the 2012 Study (response to Comment #10):

Mt. Taylor Mine
April 23, 2012 Soil Investigation
Performed by Stan Fitch, Radiation Safety Officer

Dose Rate Instrument: Eberline PRM-7 #182, BKG = 11 µrem/h average
Note: Dose rates taken at waist height (~1 meter).

Location #	Dose Rate (µrem/h)	Net Dose Rate (µrem/h)	Ra-226 (pCi/g)
MTE-1	18	7	1.7
MTE-2	15	4	0.7
MTE-3	26	15	1.4
MTE-4	24	13	1.2
MTE-5	18	7	1.4
MTE-6	15	4	1.5
MTE-7	13	2	1.5
MTE-8	14	3	2.8
MTE-9	14	3	1.8
MTE-10	14	3	1.2
MTE-11	13	2	1.2
MTE-12	13	2	2.1
MTE-13	14	3	2.7
MTE-14	14	3	1.1
MTE-15	14	3	2
MTE-16	13	2	0.8



Slope = -1.140
Y Intercept = 6.538
R^2 = 0.030
Regression test failed

Correlation

Net Dose Rate	pCi/g
0	6.54
0.25	6.25
0.5	5.97
0.75	5.68
1	5.40
1.25	5.11
1.5	4.83
1.75	4.54
2	4.26
2.25	3.97
2.5	3.69
2.75	3.40
3	3.12

Negative Correlation

5) Figure 7 has been enlarged for readability (response to Comment #11):

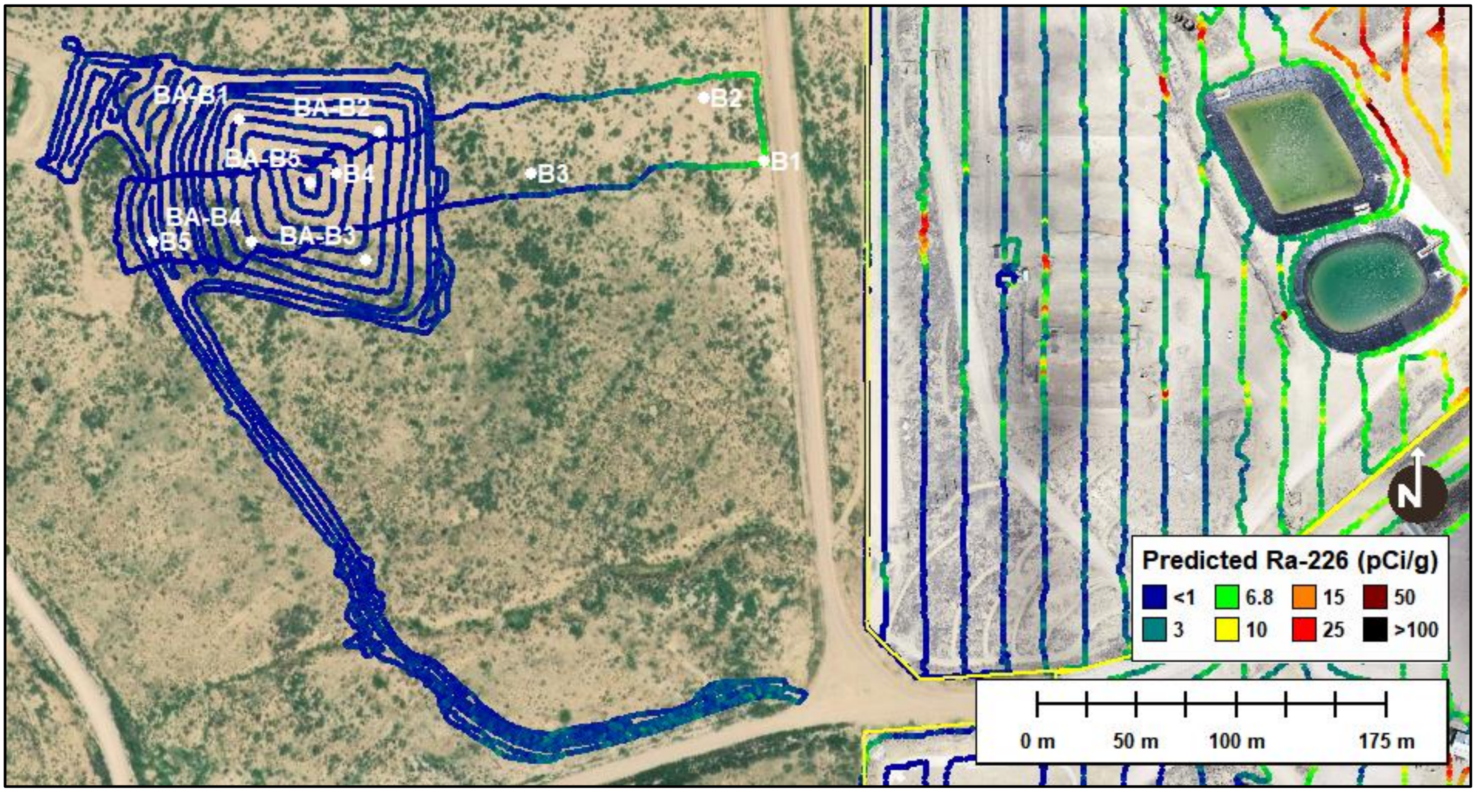


Figure 7: Gamma survey-based prediction of Ra-226 concentrations for excavated and adjacent portions of Borrow Area B, with annotated locations for confirmatory soil samples.

6) Proposed Changes to Section 4 of the Work Plan (response to Comment #13):

4 METHODS

This Section specifies the general approaches and methods that will be used to meet the following analytical objectives:

A. Characterize radiological conditions across potentially impacted land areas to identify/delineate soil impacts relative to the 17.5 µR/hr gamma cutoff goal and 24.5 µR/hr PRRL.
B. Classify survey area designations under MARSSIM guidelines (i.e. Class 1, 2, 3 or non-impacted).
C. Guide remedial excavation of impacted soils to meet the 6.8 pCi/g Ra-226 release criterion.
D. Analytical/statistical demonstration of compliance with the 6.8 pCi/g Ra-226 soil release criterion across all impacted land areas.
E. Verify compliance with the 20 pCi/m^2/s radon flux standard for the South Waste Rock Disposal Cell.
F. Verify compliance with contamination release criteria for PMLU facilities.

For the purposes of the Work Plan, the methods needed to address objectives A through C above are collectively referred to as "*Remedial Support Surveys*" (RSS), whereas those needed to address objectives D through F above are referred to as "*Final Status Surveys*" (FSS).

The basic analytical methods that will be employed under the Work Plan include the following:

1) Gamma radiation measurements (RSS excavation control) or GPS-based gamma surveys (walkover, UTV-based, or other suitable method).
2) Cross-calibration of GPS-based gamma survey data to obtain estimates of true exposure rate as measured with a high-pressure ionization chamber (HPIC).
3) Use of Site-specific gamma/soil Ra-226 correlation to predict Ra-226 concentrations based on gamma survey readings.
4) Direct soil sampling and laboratory analysis of Ra-226 concentrations as specified in Table 4 (Section 4.3).
5) Radon flux measurements over South Waste Rock Disposal Cell cover.
6) Alpha surveys for contamination in buildings, facility infrastructure, and associated equipment. As mentioned above, beta and gamma surveys may be performed to qualitatively evaluate general survey objectives.

6) Proposed Changes to Section 4, Continued (response to Comment #13), Addition of Sections 4.2 and 4.3:

4.2 Radon Flux Survey

After completion of the placement of final radon barrier (engineered soil cover) over the South Waste Rock Disposal Cell, a grid-based survey of discrete radon flux measurements will be performed in a manner consistent with EPA Method 115 as specified in applicable sections of 40 CFR 61, Appendix B (*Monitoring for Radon-222 Emissions*). The objective is to demonstrate that the average radon flux across the final radon barrier (engineered soil cover) meets the 20 pCi/m^2/s radon flux standard as specified in the MMD/NMED Joint Guidance (MMD/NMED, 2016).

4.3 Radiological Surveys of Buildings, Infrastructure, and Equipment

To support unrestricted release of mine buildings, facility infrastructure, and associated equipment to the PMLU, or release of these facilities/equipment from the Site for use by third parties or sale for scrap, systematic alpha, beta, and gamma surveys will be performed to verify that any residual radiological surface contamination does not exceed a DCGL for gross alpha activity developed to ensure that any future use of these facilities/equipment will not result in a radiological dose to any member of the public in excess of 25 mrem/yr in accordance with 20.3.4.426 NMAC as required by the mine permit (see Section 2.2 for details on release criteria).

Representative radiological survey measurements and sampling will be performed to characterize surface contamination for comparison against the total alpha release criterion specified in Section 2.2 of the Work Plan (i.e. 2,364 dpm/100 cm^2). Survey units will be comprised of a logical grouping of similar types of materials and surfaces within any given building or structure (e.g. “floor”, “Walls”, “pipes”, etc.). If the average measured surface activity in a given survey unit is below this criterion, the survey unit will qualify for release from radiological restrictions on future disposition of that portion of the habitable structure (e.g. onsite or offsite use by any future member of the public). In addition, a secondary release criterion will be that no single measurement can exceed a value 10 times the DCGL for average surface activity as specified in Section 2 of the Work Plan (i.e. 2,364 dpm/100 $cm^2 \times 10$ = 23,640 dpm/100 cm^2).[1]

In general, a combination of surface scanning and discrete measurements (static counts) will be collected to demonstrate that average levels of residual radioactivity within a survey unit are below the criteria for release. A small percentage of removeable swipe tests will be performed at representative locations to verify DCGL modeling assumptions. Volumetric sampling will be performed for materials potentially contaminated at depth into the substrate. This includes porous materials such as concrete and wood. Volumetric contamination samples will be submitted to a qualified (NELAP accredited) laboratory for analysis of Ra-226 concentration, and will be subject to chain of custody requirements.

The primary instrumentation to be used to perform surface contamination surveys is a Ludlum Model 43-93 alpha/beta scintillation detector coupled to a Ludlum Model 2360 ratemeter/scaler (or equivalent instrument pairing). A gamma detector (e.g. Ludlum Model 19) and alpha/beta tray counter (e.g. Ludlum Model 3030 or similar instrument) for removable swipe samples will also be used. All instruments will be calibrated within one year prior

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[1]This “maximum” single measurement criterion is adopted from the American National Standards Institute (ANSI) and Health Physics Society (HPS) Standard *“Surface and Volume Radioactivity Standards for Clearance*” (ANSI/HPS, 2013), the NRC’s “*Health Physics Surveys in Uranium Recovery Facilities*” (NRC, 2002) and the NRC’s “*Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, Source, or Special Nuclear Materials*“ (NRC, 1993).

to use, and function checked daily using the quality control procedures detailed in the Mount Taylor Radiation Protection Program (RPP) Manual and SOP-2 (Instrument Testing and Calibration).

The design of the contamination survey will ensure spatially representative measurements that provide adequate coverage of the survey unit. Measurement data will be averaged and results compared to the DCGL as noted above, including comparison against the maximum single measurement criterion. A FSS report will be generated for each facility survey. The FSS report will include an overview of results and detailed information regarding the number of samples taken within each survey unit, a description that includes a map or diagram showing measurement/sampling locations, tables of measurement results, and an evaluation of compliance with the release criteria. Any anomalous data will be discussed, and additional judgmental or miscellaneous sampling will be reported.

7) Proposed Changes to Section 6.1 of the Work Plan, “Sign Test” Discussion (response to Comment #18):

The Sign test compares the distribution of a set of FSS soil sampling results from the survey unit to a fixed DCGL value. The analytical result for each sample collected in the survey unit is subtracted from the DCGL, and the resulting adjusted value will have either a negative or positive sign. The statistical Sign test evaluates the proportion of adjusted values with sign that is a positive number to determine whether the median value in the survey unit is equal to or less than the DCGL. If a statistically significant proportion of adjusted values are greater than zero (i.e. the “sign” of these adjusted values are positive numbers), then the median value for the survey unit can be statistically concluded to demonstrate compliance with the DCGL. A complete description of application of the Sign test, along with several examples, can be found in Section 8 of MARSSIM (USNRC, 2000). The Sign test for FSS data will be performed with EPA’s ProUCL statistical software (USEPA, 2015) or a similar standard statistical software package.

8) Appendix A (response to Comments #2):

APPENDIX A – DERIVATION OF RELEASE CRITERIA FOR SURFACE CONTAMINATION

Derivation of Radiological Release Criteria for Facilities

Derived Concentration Guideline Levels (DCGLs) for the uranium series decay chain radionuclides (U-238 decay chain) and actinium decay series radionuclides (U-235 decay chain) were derived for radiological surface contamination on buildings, infrastructure, and associated operational equipment. A conservative residential use scenario was assumed for modeling the post-closure unrestricted use of these facilities with the RESRAD-BUILD computer code, Version 3.5 (ANL, 2009).

The “building occupancy scenario”, as described in NUREG/CR-7267 (USNRC, 2020), with associated input parameters as described in Table B-26 of NUREG/CR-7267, was used as the base case scenario for the RESRAD-BUILD model. The modeling approach was deterministic rather than probabilistic. The “building occupancy scenario” does not assume habitable use under a residential scenario, but instead assumes occupational occupancy in an office building. The building occupancy scenario uses the following assumptions per NUREG/CR-7267 (NRC, 2020):

- The building will be occupied immediately after release under the Mine Closeout/Closure Plan (RGR, 2013).
- The residual contamination is assumed to reside on inner building surfaces.
- The assumed removable fraction will be 10%.
- The exposure scenario assumes long-term chronic exposure to low-level radioactive surface contamination at levels near or below the release criteria.
- Radioactive dose results from exposure via the following exposure pathways:
    - External exposure to penetrating radiation from surface sources,
    - Inhalation of resuspended surface contamination,
    - Inadvertent ingestion of surface contamination directly from the source,
    - Inadvertent ingestion of materials deposited on the surfaces,
    - External exposure from deposited material,
    - External exposure during submersion in airborne radioactive dust, and
    - Inhalation of indoor radon aerosol.

The building occupancy scenario from NUREG/CR-7267 was modified to reflect a more residential scenario by changing select parameters from the default building occupancy scenario to those more reflective of residential occupancy, as described in NUREG/CR-7267. The following parameters which were changed from the building occupancy scenario described in NUREG/CR-7267:

- Resuspension rate (s^{-1})
- Air exchange rate (h^{-1})
- Breathing rate (m^3d^{-1})
- Indoor fraction

The resuspension rate was modified to 1.3×10^{-5} s^{-1} based on the maximum probabilistic resuspension rate described in NUREG/CR-7267 to account for more vigorous activity that may occur in the building as a residence rather than an occupational scenario. The default value from Table B-26 in NUREG/CR-7267 is 6.80E-8 s^{-1}. The breathing rate was modified to 23 m^3d^{-1} (8400 m^3y^{-1}), based on the most likely value in the RESRAD Version 6.0 default probabilistic distribution, per NUREG/CR-6697, Table 2.1 (USNRC 2000). This breathing rate is more representative of a person completing a variety of activities in the building, including sleeping, rather an occupational scenario where the critical receptor is exposed only during commercial hours of business. The indoor fraction was increased to 241 days per year (an occupancy factor of 0.66), versus a default occupancy factor for a commercial receptor scenario of

0.267. This is representative of a person spending just under 16 hours per day indoors for the entire year in the building, and is the default value for residences from NUREG/CR-7267. The air exchange rate was modified from 1.52 hr^{-1} to 0.8 hr^{-1}. While large industrial buildings generally have a greater air exchange rate, for residential use energy efficiency is commonly a desirable or necessary feature and a lower air exchange rate was conservatively assumed. An air exchange rate of 0.8 hr^{-1} is the default deterministic value (USNRC, 2020).

The following long-lived nuclides (with half-life greater than 10 days) were modeled:

- Uranium 238 (U-238),
- Uranium 234 (U-234),
- Thorium 230 (Th-230),
- Radium 226 (Ra-226),
- Lead 210 (Pb-210),
- Polonium 210 (Po-210),
- Uranium 235 (U-235),
- Protactinium 231 (Pa-231),
- Actinium 227 (Ac-227)

Each nuclide was modeled at the building occupancy scenario specified surface activity of 100 decays per minute (dpm) per square meter (m^2), which is equivalent to 1 dpm/100 cm^2. The resulting output from RESRAD-BUILD for each nuclide, is given in millirem (mrem) per year (yr^{-1}) per dpm/100 cm^2. The radiological dose-based criterion for license termination and unrestricted use from Title 20, Chapter 3, Part 4, Subpart 426.B of the New Mexico Administrative Code (NMAC) is 25 mrem yr^{-1}. This criterion is divided by each nuclide's respective output from RESRAD-BUILD to result in a DCGL for each nuclide. Uranium ore has an assumed activity fraction for each nuclide in its decay chains (uranium and actinium decay series nuclides) based on the isotopic abundance of the parent uranium isotopes U-238 and U-235 which occur in nature with natural abundances of 99.27% and 0.72%, respectively. These activity fractions are then used to derive a combined DCGL by the equation:

$$DCGL_{Combined} = \frac{1}{\sum \frac{f_N}{DCGL_N}}$$

Where:

$DCGL_{Combined}$ = The derived concentration guideline level for the mixture of radionuclides present

f_N = The activity fraction of each respective radionuclide in the mixture

$DCGL_N$ =The derived concentration guideline level for each respective radionuclide

The combined DCGL result, in dpm/100 cm^2 is then multiplied by number of emissions of alpha or beta particles from each parent radionuclide and the assumed activity of the parent radionuclide at the combined DCGL. These values are then summed. This results in separate DCGLs for gross alpha and beta surface activities.

The combined alpha/beta DCGL for the residential building scenario is 1,753 dpm/100 cm^2. The activity fractions for the combined DCGL for each decay of the long-lived parent mixture, assumes 1.35 alpha particles emitted and 1 beta particle emitted, resulting in a gross alpha DCGL is 2,364 dpm/100 cm^2, and a gross beta DCGL is 1,753 dpm/100 cm^2. For reasons described in Section 2.2 of this Work Plan, beta measurements with the instruments specified have insufficient sensitivity and are to be used only for qualitative indications of contamination, rather than quantitatively compared to a DCGL for betas.

REFERENCES

Argonne National Laboratory (ANL). 2003. ANL/EAD/03-1. User's Manual for RESRAD-BUILD Version 3. Available at: https://resrad.evs.anl.gov/docs/ANL-EAD-03-1.pdf

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