REQUEST FOR CLARIFICATIONS

RADIATION PROTECTION PLAN

Soil

The licensee addresses its previous commitments[1] for surveying subsurface soil brought to the surface during installation of injection and extraction trenches, monitoring wells, trenches for piping and utilities, etc., in Section 12.5 of the Radiation Protection Plan (RPP). The licensee stated that these soils have been previously released from license controls. However, this statement is not accurate. The three subareas remaining under NRC license SNM-928 (the license), F, G, and N, have had various surveys performed within them for the purpose of demonstrating that they could be released from radiological controls and removed from the license. These surveys are described, in part, in the licensee's May 3, 2021, letter.[2] However, there has been no regulatory action to release these subareas from the license and they are therefore not released "from license controls" as stated by the licensee.

For the purpose of surveying areas of disturbed soils (not including the 1206 Drainage sediment and spoils), the licensee divided soil depths into three groups: less than one foot below the ground surface, from one foot to one meter below the ground surface, and one meter or more below the ground surface. As the NRC staff understands the proposed method for surveying disturbed soil, the licensee will perform a gamma scan of soil before and after excavation of these three soil depths. The licensee proposed using an action level of "two times background" to determine if soil samples need to be obtained from excavated soil and sent offsite for laboratory analysis. The results for the laboratory analyses would then be compared to the requirements in License Condition 27.c. of the license to determine if the soil can be released for unrestricted use. The NRC staff notes that "2 times background" is not an objective measure and can't be used for comparison against decommissioning criteria.

Clarification requested:

Provide the rationale for dividing the soil into the three specified depths.

Provide a methodology for confirming that subsurface soil brought to the surface (not including the 1206 Drainage sediment and spoils) will not result in soils exceeding soil release criteria in any land area.

EPM Response:

As stated in the *License Transfer Order for the Cimarron Facility at Crescent, OK*, 76 FR 10918-10923, dated February 28, 2011:

> *To date, most of the decommissioning activities needed for release of the Cimarron Site for unrestricted use and to terminate Radioactive Materials License SNM–928 have been completed. The remaining activities to be completed include the release of Subareas F, G, and N as well as groundwater remediation. Groundwater contamination has been identified in Subareas F and C, as well as in the western upland and the western alluvial areas of the site.*

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[1] Agencywide Documents Access and Management System (ADAMS) Accession No. ML19154A597
[2] ADAMS Accession No. ML21123A290.

Final status surveys and confirmatory surveys have confirmed that Subareas G and N are releasable for unrestricted use, but NRC has determined that these areas should not be released until groundwater remediation is complete. Because groundwater exceeds license criteria in Subarea F, this area cannot be released for unrestricted use until groundwater remediation is complete.

The RPP inaccurately characterized Subareas F, G, and N as being released from license control. As provided in the Decommissioning Plan, there will be no excavation in any area where there is the potential for contaminated soil exceeding 30 pCi/g uranium. Accordingly, the RPP will be revised to no longer divide the soil into three specific depths and to indicate that sampling excavated soil to support decommissioning activities is not needed. No methodology is necessary for subsurface soil brought to the surface when all planned excavations will be in areas that do not exceed 30 pCi/g uranium, including the 1206 Drainage sediment and spoils. The RPP will be updated to remove the discussion of this area.

Section 12.5 will be updated to only require exposure rate surveys when non-soil material is encountered (such as discolored soil, building debris, foreign objects, etc.) as final status surveys and confirmatory surveys do not apply. The non-soil material will be removed and surveyed to determine if the material can be released for unrestricted use; if not releasable, the material will be disposed of as low-level radioactive waste. The RPP will be updated accordingly.

<u>Internal Exposure Determination</u>

NUREG-1400[3]

The licensee discusses the use of a dispersibility factor (D) less than 10. However, this is not substantiated in NUREG-1400. The factors discussed by the licensee (resin being moist, large particles, a closed processing system) are already captured by the release fraction R and the confinement factor C.

The licensee also states:

> In considering an appropriate dispersion factor for particles that do not tend to become airborne, Figure 2.6 of NUREG-1400 was reviewed. This figure provides an example where quantitative dispersion factors are determined directly through air sampling. Although quantitative values cannot be determined at this time, NUREG-1400 supports use of dispersions factors less than 1.

The NRC staff reviewed Figure 2.6 of NUREG-1400 and can't corroborate the licensee's interpretation. Figure 2.6 is related to Section 2.2.2.2 of NUREG-1400 discussing quantitative airflow studies. Figure 2.6 is related to the dispersion of aerosols from the source of a release:

[3] ADAMS Accession No. ML13051A671

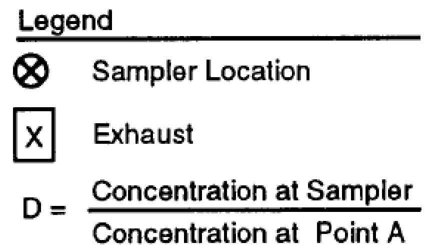

Figure 2.6. Quantitative Dispersion Factors

The “D” in Figure 2.6 is referred to as the “dispersion factor”. This is not the same as the “dispersibility factor” discussed in Section 1.2.3 of NUREG-1400 for determining potential intakes.

Clarification requested:

Provide a justification for the use of a dispersibility factor (D) less than 10 that is consistent with Section 1.2.3 of NUREG-1400 for determining potential intakes.

EPM Response:

The potential intake calculation provided in Appendix A of the RPP discusses the rationale for using a value for dispersibility of 0.1. NUREG-1400 describes dispersibility (D) in Section 1.2.3 as follows:

> *Another factor that may be appropriate to consider is the dispersibility that comes from adding energy to the system through grinding, milling, boiling, or exothermic chemical reactions. A dispersibility factor D of 10 can be applied to the calculation if cutting, grinding, heating, or chemical reactions of materials are performed.*

NUREG-1400 specifically indicates that dispersibility factor of 10 is applied to operations involving the addition of energy to the system. Processing of resin does not add energy to the system as currently designed and described in the Decommissioning Plan. As discussed in Appendix A of the RPP, a factor of 0.1 was used because, based on professional judgement, sluicing spent resin and blending it with absorbent in an enclosed low-energy blender is an operation that does not involve sufficient likelihood of dispersing the contaminant in air to consider dispersibility more than a negligible factor. If a more conservative factor of D = 1 was used to represent dispersibility as essentially a non-factor, instead of 0.1, the potential intake would be 1.1% ALI. This value still supports the air monitoring regime described in the RPP. Appendix A will be updated to incorporate D = 1, which is reasonably conservative for an enclosed system where energy is not added. This change has no effect on other provisions related to air sampling and internal dose monitoring already provided in the RPP.

10 CFR 20.1204(g).

The licensee was requested to address the requirements of 10 CFR 20.1204(g) for their expected mixture of radionuclides. 10 CFR 20.1204(g) states:

(g) When a mixture of radionuclides in air exists, licensees may disregard certain radionuclides in the mixture if—

(1) The licensee uses the total activity of the mixture in demonstrating compliance with the dose limits in § 20.1201 and in complying with the monitoring requirements in § 20.1502(b), and

(2) The concentration of any radionuclide disregarded is less than 10 percent of its DAC, and

(3) The sum of these percentages for all of the radionuclides disregarded in the mixture does not exceed 30 percent.

The licensee evaluated 10 CFR 20.1204(g)(2) and (3) but did not address10 CFR 20.1204(g)(1).

Clarification requested:

The requirements that specify when a licensee may disregard certain radionuclides in a mixture when determining internal exposure are found in 10 CFR 20.1204(g). NRC guidance on 10 CFR 20.1204(g) can be found in NUREG-1736.[4] NUREG-1736 presents several examples of calculations using mixtures of radionuclides.

The following questions and answers on the NRC website may also be helpful in addressing mixtures of radionuclides and when certain radionuclides may be disregarded when assessing internal dose:

Q121: http://www.nrc.gov/about-nrc/radiation/protects-you/hppos/qa121.html

Q403: http://www.nrc.gov/about-nrc/radiation/protects-you/hppos/qa403.html

Q453: http://www.nrc.gov/about-nrc/radiation/protects-you/hppos/qa453.html

In particular, question Q403 addresses disregarding the contribution of a radionuclide for recording and reporting purposes if it is less than 10% of the committed effective dose equivalent.

The licensee's proposed method of assessing mixtures of radionuclides does not meet the requirement of 10 CFR 20.1204(g)(1) and is not consistent with the guidance provided above for this regulation. Specifically, the licensee's proposed methodology does not account for the activity of those radionuclides it disregarded based on the derived air concentration analysis.

Either provide the NRC staff with a proposed method of assessing mixtures of radionuclides that meets the requirement of 10 CFR 20.1204(g)(1) and is consistent with the guidance provided above for this regulation or provide justification for an alternate methodology. An alternate methodology that does not meet the requirement of 10 CFR 20.1204(g)(1) will require an exemption to this regulation.

[4] ADAMS Accession No. ML013330179

EPM Response:

The licensee was requested to address the requirements of 10 CFR 20.1204(g), but §20.1204 provides requirements for “determination of internal exposure.” §20.1204(a) acknowledges that “assessing dose used determine compliance with occupational dose equivalent limits” is performed “when required under §20.1502.” The calculation provided in Appendix A of the RPP demonstrates that, as provided in §20.1502(b), no adult is likely to receive in excess of 10% of the applicable ALI in year. Updating the calculation in Appendix A of the RPP by using the more conservative dispersibility factor of D = 1 results in a potential intake of 1.1% ALI (including uranium daughters and Tc-99), which is still less than 10% of the applicable ALI in a year. Minors and declared pregnant women will not be engaged with spent resin processing activities. Appendix B provides a calculation supporting that no external monitoring is required under the provisions of §20.1502(a). As determination of internal dose is not required, the provisions of §20.1204(g) are not applicable.

The requested information is specifically related to the determination of internal dose from mixtures of radionuclides, which is not applicable at the Cimarron Site. However, the following information is provided to satisfy the NRC’s request:

Regarding the concern over mixtures of radionuclides and potential dose impact, Attachment C to Appendix A of the RPP provides a sensitivity analysis for Th-231, Th-234, and Pa-234m demonstrating that there is no need for air sampling for these radionuclides as the total quantity of unencapsulated material processed annually is less than 10,000 times the ALI. While this information by itself does not specifically address the NRC’s concern regarding dose contributions when a mixture of radionuclides in air exists, if the potential intakes are calculated (using the same assumptions used for the uranium potential intake calculation), the following potential intakes and percentages of respective ALIs results are obtained. To be complete, the calculation for the uranium isotopes, based on D = 1, and Tc-99 are provided:

Radionuclide	Potential Intake (µCi)	Fraction of ALI
U-234	3.99E-05	9.97E-03
U-235	1.50E-06	3.75E-04
Th-231	1.50E-06	2.50E-09
U-238	4.43E-06	1.11E-03
Th-234	4.43E-06	2.21E-07
Pa-234m	4.43E-06	6.32E-09
Tc-99	6.38E-04	9.11E-07

Based on the potential intake for radionuclides other than uranium, the potential dose contribution from these radionuclides are several orders of magnitude less than any dose received from an intake of uranium. The potential intake from uranium exposure (sum of the fractions) is 1.14% ALI. The sum of the fractions including the uranium daughters and Tc-99 is 1.14% ALI. Accordingly, the requirement in 10 CFR 20.1204(g)(1), if it applied, is satisfied; specifically monitoring requirements under §20.1502(b) is not required and is unnecessary for demonstrating compliance with the dose limits in §20.1201.

Radiation Protection Instrumentation

Scan MDC

The licensee provided an example scan MDC calculation for beta contamination. However, the scan MDC for alpha contamination uses a different equation. The licensee should provide an example of a calculated scan MDC for alpha contamination as previously requested.

The NRC staff notes that there appears to be an extra subscript for the surface efficiency in its equation for scan MDC:

$$Scan\ MDC = \frac{d' \times \sqrt{b^i} \times \left(\frac{60}{1}\right)}{\sqrt{p} \times \varepsilon_i \times \varepsilon_{is} \times \left(\frac{probe\ area}{100}\right)}$$

Where:

d' = the index of sensitivity, typically 1.38 is based on 95% correct detections and 60% false positive rates but may be modified by the RSO depending on project decision as to confidence desired in ability to detect elevated area
b_i = background counts in the observation interval
i = observational interval (in seconds), based on the scan speed and areal extent of the contamination, 1 second is used.
ε_i = the instrument efficiency
ε_s = the surface efficiency
probe area = physical probe area of the radiation detector (cm^2)
p = surveyor efficiency, 0.5

This appears to have been addressed in the RPP (Note 3 in Section 7.5 of the RPP). However, in Note 1, the licensee provides the following equation for minimum detectable activity (MDA), where PA is the probe area as defined above:

$$MDA = \frac{3 + 3.29\sqrt{R_b T_s (1 + {T_s}/{T_b})}}{(E \times T_s)/(\frac{P_A}{100})}$$

The PA/100 term appears to be incorrectly divided in the denominator.

In Section 7.5.1, Contamination Survey Instrument Counting Efficiency, of the RPP, the licensee states:

> The RSO may provide alternative methods for determining calibration efficiency, which could include use of the counting efficiency provided on the instrument calibration certificate. Alternative methods shall be documented with the instrument records.

Please provide the NRC staff with an example of such an alternative method. The NRC staff can accept alternative methods to those given in guidance documents, but not without details sufficient for the NRC staff to understand those alternative methods.

Also, please clarify the use of “calibration” efficiency and “counting” efficiency as used above.

<u>Clarification requested</u>:

- Provide an example of a calculated scan MDC for the alpha contamination expected at the Cimarron site as previously requested.

<u>EPM Response:</u>

NUREG-1575, Rev. 2, provides Equation 6-14 for determining the scan MDC for

$$Alpha\ scan\ MDC\ = \frac{[-\ln(1 - P(n\ \geq 1)] \times\ 60/i}{\varepsilon_t\ \times\ \frac{W}{100}}$$

Where,

P(n ≥ 1) = 90% probability of detecting 1 count (per NRC reference in the Request for Clarification)

i is the observation interval (in seconds), which is determined by dividing the width of the detector in the direction of the scan in centimeters divided by the scan speed in centimeters/second

ε_t is the detector efficiency (4π)

W is the physical probe area (cm^2) [incorrectly referred to as “*A*” in NUREG-1575]

For the Ludlum Model 2360 equipped with the Ludlum 43-93 probe, the probe width is 9.14 cm and the procedural scan speed is one probe width per second, 9.14 cm/s. This results in *i* = 1 s. The 4π efficiency for the 43-93 probe used at the Cimarron Site is 17.6%. The probe area is 100 cm^2. Substituting these values into Equation 6-14, the typical *Alpha Scan MDC (90% detection probability)* for this instrument used at the Cimarron Site is 785 dpm/100 cm^2.

- Confirm the PA/100 term in the MDA equation above.

<u>EPM Response:</u>

The two typographical errors noted by the NRC staff for the efficiency subscript (ε_{is}) and division by the term, P_A(probe area)/100, will be corrected in the RPP.

- Clarify the use of “calibration” efficiency and “counting” efficiency as used above.

<u>EPM Response:</u>

Section 7.5.1 of the RPP incorrectly used the term “calibration efficiency” when the term, “counting efficiency” should have been used. This editorial change will be made to the RPP.

- Provide the NRC staff with the details of any alternative methods that the licensee wants to use for determining counting efficiency, scan MDC, etc.

EPM Response:

As discussed in the section 7.5.1 of the RPP, the RSO may approve alternate methods for determining counting efficiencies for instruments used at the Cimarron Site. Currently, as reflected in the RPP, two alternatives are currently addressed; using the efficiency determined by the calibration vendor or using a site-specific value. Based on the recommendation of the instrument manufacturer, who is the calibration vendor, a site-specific counting efficiency is calculated when an instrument returns from being calibrated. The site-specific counting efficiency is determined by counting the site's certified sources (previously discussed in response to the NRC staff's previous Requests for Clarification on the RPP) to determine the counts per disintegration (i.e., counting efficiency).

Surface efficiencies

The licensee provided a novel approach to assigning beta surface efficiencies to its radionuclide mixture. This novel approach derived an overall surface efficiency for the radionuclide mixture based on an average beta energy of that mixture.

The NRC staff can't corroborate this novel approach as it doesn't address the physics behind the derivation of the surface efficiency. The surface efficiency is derived for individual particles, not for mixtures. For example, a radionuclide with a high energy beta particle emission, such as Pa-234m (E_{max}=2.28 MeV), has no bearing on a radionuclide with a low energy beta particle emission, such as Th-234 (E_{max}=0.19 MeV).

Clarification requested:

The licensee should provide a calculation of counting efficiencies that utilizes an accepted methodology for incorporating surface efficiencies. If the licensee wants to use an alternative approach to determine counting efficiency, it should provide sufficient information for the NRC staff to evaluate the approach. This information could include, for example, a comparison of the alternative approach with the accepted methodology and any empirical data developed by the licensee.

EPM Response:

The weighted surface efficiency for the beta emitters is calculated in the following table for 1% and 4% enriched uranium as an example. The results indicate that the surface efficiency for beta at the Cimarron Site should be 0.3. Currently, the Site uses a beta surface efficiency of 0.5 as indicated in the note following section 7.5.1 of the RPP. The note will be updated to reflect a beta surface efficiency of 0.3.

Table: Weighted Surface Efficiency for Beta Emissions

		1% LEU					
Radionuclide	Max Beta Energy (MeV) (Note 1)	Fraction in Mixture (Note3)	Normalized Fractional Distribution of Mixture	Instrument Efficiency (Note 1)	Weighted Instrument Efficiency	Surface Efficiency	Weighted Surface Efficiency (Note 4)
U-234		0.548					
U-235		0.028					
U-238		0.425					
Th-231	0.283	0.028	0.015	0.240	0.004	0.250	0.003
Th-234	0.178	0.425	0.226	0.170	0.038	0.250	0.037
Pa-234m/ Pa-234	2.24	0.425	0.226	0.380	0.086	0.500	0.165
Tc-99 (Note 2)	0.294	1.000	0.532	0.250	0.133	0.250	0.127
				Total	0.26	Total	0.33
		4% LEU					
Radionuclide	Max Beta Energy (MeV) (Note 1)	Fraction in Mixture (Note3)	Normalized Fractional Distribution of Mixture	Instrument Efficiency (Note 1)	Weighted Instrument Efficiency	Surface Efficiency	Weighted Surface Efficiency (Note 4)
U-234		0.793					
U-235		0.044					
U-238		0.163					
Th-231	0.283	0.044	0.032	0.240	0.008	0.250	0.008
Th-234	0.178	0.163	0.119	0.170	0.020	0.250	0.020
Pa-234m/ Pa-234	2.24	0.163	0.119	0.380	0.045	0.500	0.088
Tc-99 (Note 2)	0.294	0.793	0.730	0.250	0.182	0.250	0.178
				Total	0.26	Total	0.29

Note 1	Source: Table A-30 of NUREG-1507
Note 2	Tc-99 set equal to uranium activity
Note 3	Isotopic distribution of uranium mixture from "A Review and Verification of thee Isotopic Distribution of Enriched Uranium and the Impact on Decommissioning Considerations", A. J. Nardi, Tracy Chance and John F. Conant, presented at the HPS 2007 Midyear Topical Meeting, Jan. 21-24, 2007; Secular Equilibrium assumed for progeny
Note 4	E. W. Abelquist, Decommissioning Health Physics, A Handbook for MARSSIM Users, Second Edition, Case Study, pp 288-290