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LOST CREEK ISR, LLC

August 5, 2009

Ms. Tanya Palmateer Oxenberg
Project Manager
U.S. Nuclear Regulatory Commission
Mail Stop T8F5
Two White Flint North
11545 Rockville Pike
Rockville, MD 20852-2738

**Re: Lost Creek Project Responses to Health Physics Comments
Docket No. 40-9068
TAC No. LU0142**

Dear Ms. Oxenberg,

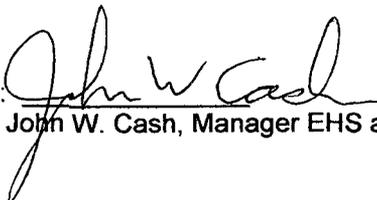
Please find behind this cover, in duplicate, Lost Creek ISR, LLC's responses to the health physics comments posed by the NRC during the April 23, 2009 teleconference.

Lost Creek ISR, LLC has worked diligently with several experienced staff and contract Health Physicists to develop what we believe are appropriate responses based on existing law, guidance and long-standing, accepted practices. We look forward to continuing to work with NRC staff to resolve these few remaining issues.

If you have any questions regarding this submittal, please feel free to contact me at the Casper office.

Regards,

Lost Creek ISR, LLC
By its Manager, Ur-Energy USA Inc.

By: 

John W. Cash, Manager EHS and Regulatory Affairs

Cc: Nancy Fitzsimmons – Ur-Energy USA Inc., Littleton

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NRC and Lost Creek ISR, LLC's Detailed Discussion Summary

1) U₃O₈ / Accident scenarios

RAIs: 4.1(5)

Discussion (23 APR 2009):

In its analysis of a thickener failure and spill, NUREG/CR-6733 analyzed the material U₃O₈. Lost Creek ISR, LLC (LCI) relied on this analysis for comparison to its uranium recovery operations. However, LCI's process will produce uranyl peroxide (generally, UO₄•2H₂O), not U₃O₈.

Lost Creek Response (23 APR 2009):

LCI responded that it thought it had answered the question because the terms U₃O₈ and uranyl peroxide are used interchangeably. According to LCI, U₃O₈ does not exist in nature. LCI requested guidance regarding a response that would satisfy the NRC Staff.

NRC staff disagreed with characterizing U₃O₈ and uranyl peroxide as interchangeable compounds. NRC staff stated that it is looking to see how substituting uranyl peroxide affects the accident scenario. This is necessary because U₃O₈ and uranyl peroxide have different chemical and physical properties that may impact the dose calculations.

Action:

LCI will review the accident analysis in NUREG/CR-6733 and state how their process is or isn't bound by that analysis.

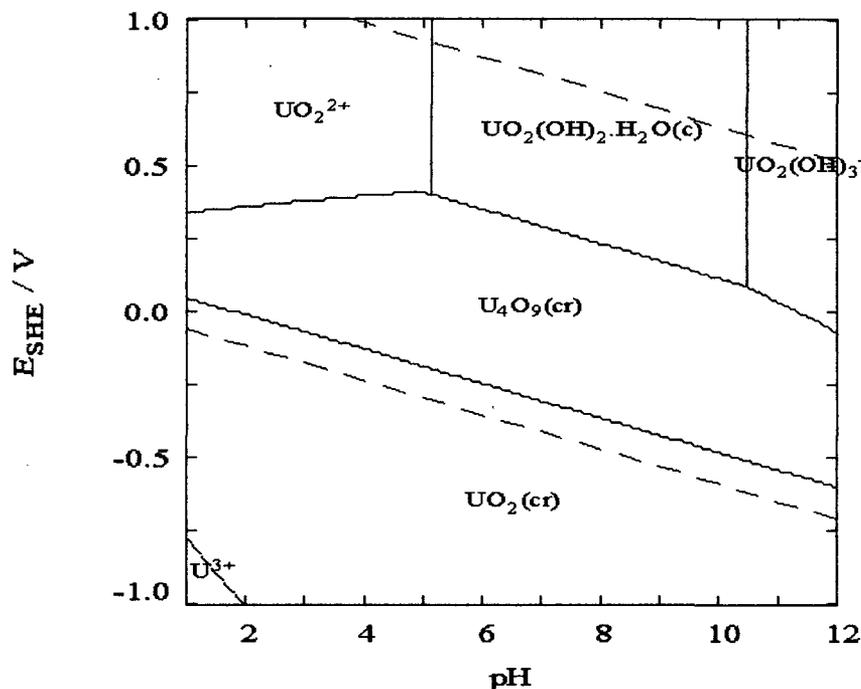
August 5, 2009 Response:

With regard to this accident scenario, NUREG/CR-6733 Section 4.2.1 assumes insoluble uranium, the worst-case assumption. If the material involved in the accident were more soluble, the dose to a worker on site would be reduced by the ratio of the more soluble annual average DAC to the insoluble DAC. The dose to a member of the public would be reduced by the ratio of the annual average effluent release limit for the more soluble uranium to the effluent limit for insoluble uranium. This is quantified below.

The products of interest are uranyl peroxide (UO₄) and or uranyl trioxide (UO₃) and /or their hydrates (not "U₃O₈") as a direct result of the elution and precipitation chemistry to be used and since even when drying is conducted, the Lost Creek ISR product will be dried with a low temperature vacuum dryer. These products are historically considered much more soluble than U₃O₈. A detailed discussion of the relatively solubility of these and related industrial uranium products as described in the literature over the last 30+ years is provided in response to Item # 2.

Uranium exists in various oxide compounds depending upon the Eh and pH of the system as shown in the Pourbaix diagram below:

$$[U]_{TOT} = 10.00 \mu\text{M}$$



The Pourbaix diagram for uranium in a non-complexing aqueous medium (eg perchloric acid / sodium hydroxide).¹²

Pourbaix, M., Atlas of electrochemical equilibria in aqueous solutions. 2d English ed. 1974, Houston, Tex.: National Association of Corrosion Engineers.

This diagram is for 25 degrees Centigrade and shows the variety of compounds present at varying Ehs and pHs.

When eventually dried by a modern vacuum drier at relatively low temperature no "U3O8" will be produced. Again, hydrates of UO_3 and UO_4 are expected. Products shipped from uranium recovery facilities vary in color from yellow to orange yellow to dark green (or even brown or black) depending on the water of hydration and oxygen content of the material. These variations are caused by differing temperatures of drying (rotary vacuum versus calcining) and different methods of precipitation, (hydrogen peroxide, ammonia etc.). It has been demonstrated that these color variations represent differences in chemical composition and therefore relative solubility (see references presented for Issue #2). In general, the darker the color, the lower the uranium valence (+ IV, e.g. UO_2 thru + VI, UO_3) and the more "insoluble" is the product. Additional assumptions stated in the thickener failure and spill scenario of NUREG CR-6733 also would tend to "maximize" dose to both workers and public relative to more realistic and credible emergency response circumstances at the Lost Creek ISR. This comparison is summarized in Table 1 below:

Table 1: Comparison of Assumptions in the NUREG /CR-6733 Accident Scenario for Thickener Failure and Spill (Section 4.2.1) vs. More Realistic LCI Emergency Response Assumptions

NUREG/CR -6733 Assumption	LCI Most Credible Case	Impact on Worker Dose*	Impact on Public Dose*
Product is insoluble U308, ICRP 19 Class Y / ICRP 66 Class S**	Product is relatively soluble U04 and/or U03 hydrates - ICRP 19 Class D or W / ICRP 66 Class F or M	DAC (Y) = 2E-11; DAC (W) = 3E-10 Therefore Class W dose = 15% of Class Y dose	Effluent concentration (Y) = 9 E-14; (W) = 9 E-13 Therefore Class W dose = 10% of Class Y dose
Design features (berms, sumps) at thickener inadequate to contain entire thickener contents; 20 % escapes building	Berms are designed to contain at least the volume of the two largest tanks combined	Cleanup within a building equipped with berms, sumps, and wash-down water minimizes cleanup time and exposure	Spill contents remain within building thereby virtually eliminating the potential for wind blown particulate
Takes no credit for "immediate" emergency response actions, assumes entire volume dries and is available for dispersion	Plant alarms and/or observation would alert staff to occurrence of event quickly; clean up actions would be initiated before majority of volume can dry including wetting / wash down techniques to move spilled material to bermed areas and sumps and other wet collection methods	Much less source term available (lower release fraction) for dispersion and therefore less dose	Much less source term available (lower release fraction) for dispersion and therefore less dose
Takes no credit for use of PPE by clean up workers	Workers involved in clean up of spilled material would be wearing respirators in accordance with an approved respiratory protection program per, e.g., 10 CFR 20, Subpart H	Dose assignment can be reduced by appropriate protection factor for device(s) used	None
Takes no credit for emergency response planning, procedures and associated training	Response to spill would be conducted in accordance with previously developed and approved emergency response protocols. Minimizes time to respond; equipment needed readily available; enhances efficiency of worker performance to affect clean up due to in place emergency response procedures, exercises and training.	In place and exercised emergency response procedures, readily available equipment and trained workers will reduce worker dose	In place and exercised emergency response procedures, readily available equipment and trained workers will reduce source term and therefore offsite dose to public

* Derived air concentrations (DAC) for workers and effluent concentrations released to unrestricted areas from 10 CFR 20, App B, Tables 1 and 2 respectively; units in uCi/ml. Although the products of interest are likely to be TGLD Class D (see discussion in response to issue 2), Class W is conservatively assumed.
 ** Task Group on Lung Dynamics, ICRP 19, Metabolism of Plutonium and Other Actinides, (1974); ICRP 66 Human Respiratory Tract Model for Radiological Protection (1994).

NUREG CR-6733 Figure 4.2, reproduced below, indicates all doses to members of the public are well below any applicable standards, and would be further reduced to about 1 mrem / year if just solubility alone were considered and less if other factors presented in Table 1 were taken into account.

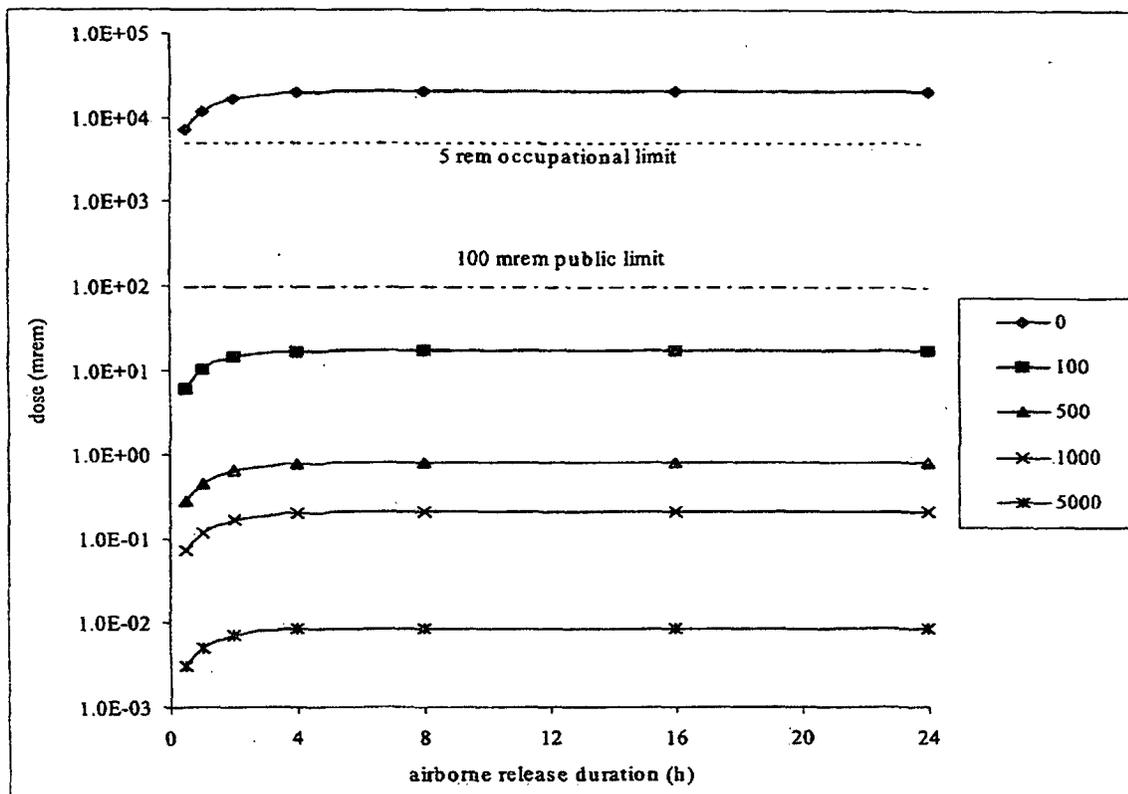


Figure 4-2. A plot of the downwind doses at various x-distances (meters) ($y = 0, z = 1$ m) from a U_3O_8 spill, based on different airborne release durations (length of time that the U_3O_8 spill receives no mitigating action after drying to a point when airborne release is possible). These dose estimates assume that no remedial or personnel protection actions are implemented.

With regard to potential on-site (occupational) doses, the analysis in NUREG/CR-6733 for an accident involving thickener failure shows a potential dose at the center of the spill, to someone standing in the spill for four hours after the spill had dried, could exceed 5 rem. The assumption is that the spill consists of insoluble Class Y U_3O_8 . Reducing the dose estimate based on solubility considerations alone (ratio of DACs - see Table 1) results in a worker dose projection of about 500 mrem, not taking into account other credible emergency response mitigating actions as presented in Table 1. On page 4-22 of NUREG /CR-6733 it is stated that, "It is reasonable to assume that cleanup personnel would be outfitted with protective equipment including respirators." It is also likely that any spill would be cleaned up before it dried. The implication is that the dose is minimal while the spill is wet. Maintaining the spill "wet" during cleanup is an expected method of collection which will ensure dust control and minimize exposure potential. Air sampling during the cleanup process will provide a record of actual exposure.

2) Derived airborne concentration (DAC)

RAIs: 4.1(2), 5.7.3(1), 5.7.4(5), 5.7.5(1)

Discussion (23 APR 2009):

NRC staff stated that it could not find justification for assuming that the use of Class D for calculating internal dose adequately represents the inhalation class of uranium compounds that could be encountered at the proposed facility. This issue is being addressed at all facilities, proposed and existing.

Lost Creek Response (23 APR 2009):

LCI explained that it used Regulatory Guide 8.30 and previous experience at licensed uranium recovery facilities to derive an inhalation class for its operations. Further, since the it (sic) was not drying its product, LCI did not expect the solubility to change because uranyl carbonate and uranyl peroxide are expected to be soluble or Class D.

NRC staff stated that Regulatory Guide 8.30 did not provide specific guidance on which inhalation class should be applied to uranium recovery operations, other than to consider yellowcake "soluble" if dried at low temperatures. However, this terminology does not comport with the current regulatory basis of 10 CFR 20, Appendix B, which uses a three-tiered system of inhalation classes; D, W, and Y. Furthermore, the regulations do not specifically address the carbonate and peroxide forms of uranium that are relevant to LCI's operations.

LCI questioned the staff on how to derive an inhalation class for unlisted materials.

NRC staff responded that LCI could make conservative assumptions to begin operations and that, once operating, it could use site specific data to derive an inhalation class (or combination of classes) that is more representative of its operating conditions.

LCI is concerned that the Regulatory Guides are incorrect and that this is complicating the review.

Action:

LCI will research the issue and get back to us. The Path forward is to provide a justification for the proposed inhalation class.

August 5, 2009 Response:

Uranium will be present at the facility exclusively in relatively soluble forms i.e., uranyl carbonates, (various forms) uranyl trioxide (UO₃), uranyl peroxide (UO₄) and hydrates of UO₃ and UO₄. The lixiviant uses oxygen and carbonate to dissolve and mobilize the uranium minerals in situ. Accordingly, the uranium goes into solution as a carbonate. If the uranyl carbonates formed were not very soluble, the in situ mining process could not work. The exact species of carbonate is largely dependent upon the pH of the solution. When the pH is from 5.1 to 6.6 uranyl monocarbonate (UO₂CO₃) will be dominant. When the pH is from 6.61-to 8.0 uranyl bicarbonate [UO₂(CO₃)₂]⁻² will dominate. When the pH is greater than 8.0 uranyl tricarbonate [UO₂ (CO₃)₃]⁻⁴ will dominate. There is overlap between each of these pH ranges so it is likely that any two species could exist at the same time. The lixiviant pH will be slightly basic so we would expect a combination of uranyl bicarbonate and uranyl tricarbonate. The

uranyl carbonates will maintain their identity when they adsorb to the resin and throughout the elution. However, when acid is added to the precipitation cell the carbonate complexes are destroyed and disassociate to form uranyl ions.

When hydrogen peroxide is added to the precipitation vessel the uranium is oxidized further to form uranyl peroxide ($\text{UO}_4 \cdot n\text{H}_2\text{O}$). This is the final chemical reaction to be considered under this licensing action. When eventually dried by a modern vacuum drier at relatively low temperature, a combination of UO_4 and UO_3 and their hydrates will result. No " U_3O_8 " can be produced since the combination of 3 uranium to 8 oxygen atoms is not possible given the valence states that are available for these elements under the thermal conditions of low temperature vacuum driers.

Although specific studies and references on solubility (e.g., in vitro solubility studies in simulated lung fluids, historical animal studies etc) for UO_4 are sparse (a few specific references are provided below), numerous references appear in the literature over 30 + years regarding general solubility characteristics of industrial uranium compounds (representative list also provided below). The UO_4 product should be Task Group on Lung Dynamics (TGLD - ICRP 19) class D or W (most or moderately soluble), which is equivalent to ICRP 66 class F or M (fast or medium dissolution). NRC staff suggests justification is not adequate to assign any class other than insoluble class Y as would be appropriate for high-fired U_3O_8 and that "Regulatory Guide 8.30 does not provide specific guidance. ...other than to consider yellowcake soluble if dried at low temperatures". The issue of assumed solubility class is critical in establishing the appropriate DAC for defining air-monitoring parameters, for worker airborne exposure control and dose assessment.

The following points support a Class D or W designation for UO_4 :

- RG 8.30 in fact calls out UO_4 specifically: "Yellowcake dried at low temperature, which is predominantly composed of ammonium diuranate, or in the new processes uranyl peroxide, both are more soluble in body fluids than yellowcake dried at higher temperature; and a relatively large fraction is rapidly transferred to kidney tissues"(Refs. 9 to 11)". Note that these references are included in the general list below.
- Reference: *Proposed Standards for Acute Exposure to Low Enriched Uranium for Compliance with 10 CFR 70.61*, Kathren R.L and Burklin R.K., Operational Radiation Safety, V. 95.2. August 2008 Page S123 – "...the more soluble compounds of uranium such as... and UO_4 are more quickly absorbed into the blood and therefore exhibit toxic effects in moderate doses (ASTDR 1999, Stannard 1988). Note that these references are also included the general list below.
- Personal Communication with Ron Kathren, PhD, CHP. Ron has been considered for many years one of the health physics profession's leading experts on uranium toxicity and metabolism. In a recent email to Steve Brown, CHP, SENES Consultants Ltd., regarding the question of UO_4 solubility, Ron responded as follows: " UO_4 is generally considered to be relatively soluble (I would use the ICRP classification here) with chemical toxicity predominant at low enrichments (say below about 15%). Chemically, once the U is absorbed into the body it behaves exactly the same as uranium from any other uranium compound, and the ICRP biokinetic model, albeit admittedly imperfect, is probably your best bet to describe the behavior of an intake. What Rich and I put in our articles is clearly applicable to UO_4 ". (NOTE: See the Kathren and Burklin reference above)

- Reference: Solubility Characteristics of Airborne Uranium From an In Situ Uranium Processing Plant. Metzger R, Wichers D. et al. Health Physics 72.3, March 1997 p 418. Results indicated airborne U in wet process area = 97% with dissolution T1/2 = 0.3 days; airborne U in drum load out area = 97% with dissolution T1/2 = 0.25 days. These results are clearly indicative of a TGLD Class D or ICRP 66 Class F compound. See ICRP 19, Task Group on Lung Dynamics *Metabolism of the Compounds of Plutonium and Other Actinides* (1974) and ICRP 66 *Human Respiratory Tract Model for Radiological Protection* (1994).

Note: We understand that the NRC Staff does not consider this paper as being an acceptable source of information on UO₄ solubility. This is of concern in that this is a peer reviewed scientific paper published in the Journal of Health Physics, the premier publication of the Health Physics Society. This study was undertaken by a licensee when 10 CFR 20 was revised to include solubility classes, and presents actual data from an operating facility replicating essentially the same experimental protocols as historical methods reported in the literature for determining solubility of uranium mill products (see specifically references below #s 1,2,3,6,7,8,12,13). We respectfully request an explanation of the basis for the NRC Staff questioning the credibility of this study.

Examples of some additional studies and references published over the last 30 + years that specifically address solubility and solubility class of uranium mill and related uranium fuel cycle uranium compounds are provided below:

1. *Preliminary Study of Uranium Oxide Dissolution in Simulated Lung Fluid*. R.C. Scipsick, et al, Los Alamos National Laboratory report LA – 10268-m, UC-41, Jan, 1985
2. *The Solubility of Some Uranium Compounds in Simulated Lung Fluid*, N. Cook and B Holt, Health Physics 27, 69-77, 1974
3. *In Vitro Solubility of Yellow Cake Samples from Four Uranium Mills and Implications for Bioassay Interpretation*, A. Eidson and J. Mewhinney, Health Physics 39, 893-902, 1980
4. *Toxicological profile for uranium (Update)*. Prepared by Research Triangle Institute for U.S. Department of Health and Human Services, Public Health Service, Agency for Toxic Substances and Disease Registry. September 1999.
5. *Biokinetics model for uranium inhalation/excretion of uranium mill workers*. Alexander R.E In: Moore RH, Ed. Biokinetics and analysis of uranium in man. United States Uranium Registry Report USUR-05, HEHF-47, 1984.
6. *Dissolution Fractions and Half Times of Single Source Yellowcake in Simulated Lung Fluids*. M. Blauer, J Kent and N Dennis, Health Physics 42, 469-477, 1982
7. *Characterization of Yellowcake and Implications for Uranium Mill Bioassay*. S Brown and M. Blauer, proceedings of Conference on Analytical Chemistry and Bioassay, Ottawa, October, 1980
8. *Physical and Chemical Parameters Affecting the Dissolution Characteristics of Yellowcake in Simulated Lung Fluids*. M. Blauer and S. Brown, Abstracts of the 25th Annual Meeting of Health Physics Society, Seattle, Paper # 177, Pergamon Press 1980
9. *Biokinetics and Analysis of Uranium in Man*. Proceedings of Colloquium held at Richland, Washington, August, 1984, United States Uranium Registry, R Moore ed., USUR – 05 HEHF-47
10. *Applications of Bioassay for Uranium*, R.E Alexander WASH-1251, U.S. Atomic Energy Commission, Washington, DC, 1974.
11. *Analysis of Uranium Urinalysis and In Vivo Measurement Results from Eleven Participating Uranium Mills*. Spitz, H. B., J.C. Simpson, and T. L. Aldridge, NUREG/CR-

2955 Pacific Northwest Laboratory, Battelle Memorial Institute for U.S. Nuclear Regulatory Commission 1984

12. *Solubility Classification of Airborne Products from Uranium Ores and Tailings Piles.* D. R. Kalkwarf. NUREG/CR-0530, USNRC 1979

13. *In Vitro Dissolution of Uranium Product Samples from Four Uranium Mills.* F. Eidson and J.A. Mewhinney NUREG/CR-0414, USNRC 1978

Conclusion

Accordingly, although UO₄ has been shown to be a TGLD (ICRP 19) Class D or ICRP 66 Class F compound, we will assume it to be Class W / Class M for purposes of establishing the initial DAC upon plant startup. Studies on Lost Creek products involving compound identification through x-ray diffraction analysis or other crystallographic analysis and/or dissolution studies in simulated lung fluids may be performed in accordance with the established protocols (well documented in the literature – examples above) to establish if Class D / Class F may be more appropriate. Upon initiation of product drying, similar studies may be performed to assess the chemical composition and solubility characteristics of the "UO₄/UO₃ product". This is appropriate to define not only the relevant DAC, but also the appropriate sampling frequencies and action levels for the plant uranium bioassay program.

3) Worker dose calculations

RAIs: 5.7.4(4)

Discussion (23 APR 2009):

Industry practice has been that the plant air particulate samples would be analyzed for gross alpha activity but assumed to be primarily, if not all, due to natural uranium. However, NRC's regulations in 20.1204(g) are specific with respect to mixtures of radionuclides. Radionuclides may only be disregarded if certain criteria are met. Otherwise, doses from individual radionuclides must be addressed. The licensee, therefore, must characterize the radionuclides in the plant or apply the gross alpha activity to the radionuclide with the most restrictive DAC (10 CFR 20.1204(f)). In this case, since thorium is in the process stream, the DAC for thorium would be the controlling radionuclide.

Lost Creek Response (23 APR 2009):

LCI responded that such characterization is not possible until the facility has been in operation.

NRC staff stated that LCI could use conservative assumptions to estimate doses for the initial start of operations and that licensees are required to make appropriate surveys to evaluate the concentrations of radioactive materials and potential radiological hazards (10 CFR 20.1501). In response, LCI stated that it will make a set of assumptions for initial startup and then it will develop methods for determining actual isotopic concentrations and refining the DAC if necessary.

Action:

LCI will submit a plan that is consistent with its response above.

August 5, 2009 Response:

LCI recognizes that the conditions specified in 10 CFR 20.1204(f) must be addressed to satisfy the requirements of 10 CFR 20.1501. However, we believe that the exceptions specified in 10 CFR 20.1204(g) are directly applicable to ISRs as the discussion below indicates.

Although 20.1204(f) requires that in situations where the identity of each radionuclide in a mixture is known, but the concentration of one or more of the radionuclides in the mixture is not known, the DAC for the mixture must be the most restrictive DAC of any radionuclide in the mixture. However, per 20.1204 (g) licensees may disregard certain radionuclides in the mixture if the licensee uses the total activity of the mixture and the concentration of any radionuclide disregarded is less than 10 percent of its DAC, and the sum of these percentages for all of the radionuclides disregarded in the mixture does not exceed 30 percent. These three conditions are met at ISRs as the discussion below demonstrates.

Isotopes of thorium are not a major component of the dose at uranium in-situ recovery facilities since very little thorium is mobilized from the host formation by in-situ uranium recovery operations. Examples supporting this include the following:

Studies performed in the late 1970s and early 1980s of radionuclide mobilization from three production scale ISRs and several R & D plants (see references 1-3 below) indicated a relatively small portion of the uranium daughter products in the ore body are actually mobilized by the lixiviant. The vast majority of secular equilibrium radionuclides remain in the host formation. The table below presents typical concentration ranges for the facilities studied in the

processing plant feed stream (pregnant lixiviant) as well as the reformed tails (barren lixiviant) being returned to the ore body. Thorium 230 appeared to equilibrate and very little was actually removed by the process. The majority of the mobilized radium 226 (80—90 percent) which was estimated to be 5~15 percent of the calculated equilibrium radium in the host formation, followed the calcium chemistry in the process and resulted in radium carbonates / sulfates in the calcite byproduct waste streams. Little, if any Lead 210 was mobilized as the lead carbonate complexes formed in situ are virtually insoluble in the lixiviant processes studied.

Radionuclide Concentrations in Process Streams (Bq/l)

	U ₃ O ₈	Th 230	Ra 226	Pb 210
Pregnant Lixiviant (produced water)	1,500 - 6,300	56-93	10 - 150	<1
Barren Lixiviant (injected water)	20 - 30	48 -81	1.9 - 4.4	<1

1. Brown, S. 1982, *Radiological Aspects of Uranium Solution Mining*, In: Uranium, 1, 1982, p37-52, Elsevier Scientific Publishing Co.
2. Brown, S, 2007, *Radiological Aspects of In Situ Uranium Recovery*. American Society of Mechanical Engineers, Proceedings of 11th International Conference on Environmental Management, Bruges, Belgium; ASME Press, New York, NY, ISBN 0-7918-3818-8
3. Brown, S, 2008, *The New Generation of Uranium In Situ Recovery Facilities: Design Improvements Should Reduce Radiological Impacts Relative to First Generation Uranium Solution Mining Plants* (In press). International Atomic Energy Agency, "Low environmental impact uranium mining and remediation:15 years of multinational experience through Uranium Mine Remediation Exchange Group", IAEA- TECDOC-Number to be assigned, IAEA, Vienna , (2009)

Additional analysis of injection solutions and air particulate samples at ISR facilities in the past has shown that thorium-230 is not present in any appreciable concentrations and that this is the basis for the use of the DAC for natural uranium. In addition to the fact that very little thorium is mobilized in situ, the ion exchange (IX) resin used in ISR facilities is specific for removal of uranium. Thorium compounds are not removed by the IX resin and are therefore not present in the process downstream of the IX columns (e.g., elution, precipitation, and drying circuits).

According to NUREG CR-6733 (*A Baseline Risk - Informed Performance - Based Approach for In Situ Leach Uranium Extraction Licenses*, USNRC 2001), thorium is not in the mixture of radionuclides present at ISR facilities. NUREG CR-6733 Table 4-5 lists assumed activities used for pregnant lixiviant and loaded resin spill scenarios and neither matrix contains thorium. In fact for all radionuclides listed in this table as components of the pregnant lixiviant, with the exception radon 222 gas, the assumed relative activities are all < 3 % of the uranium values.

Conclusion

Based on the studies and reports in the professional literature as described above, the exclusions of 20.1204 (g) clearly apply to ISRs. Historical evidence indicates thorium does not need to be considered in air particulate sampling analysis and in establishing an appropriate

DAC at an ISR since it will essentially be absent ($\ll 10\%$) from the radionuclide mixture. Similarly, since only a small percent of the equilibrium radium in the host formation is mobilized, a similar conclusion applies with regard to radium's contribution to the DAC. Therefore, it must additionally be true that the total contribution of radionuclides in the mixture other than uranium for which the 20.1204 (g) exclusion applies is $< 30\%$. Accordingly, traditional practice throughout the ISR industry for > 30 years of analyzing air samples via gross alpha counting using the relevant U natural DAC (per solubility class) is technically appropriate, protective of workers and provides an appropriate "standard of care".

Since pregnant lixiviant is the initial source of all airborne particulate, a sample of pregnant lixiviant will be analyzed upon facility start-up to verify the expected concentration of radionuclides. Additional samples will be collected after any significant process change that may impact lixiviant chemistry.

4) Contamination control program

RAIs: 5.7.6(1) & (3)

Discussion (23 APR 2009):

Issue 1

On page 5-40 of the Technical Report, first full paragraph, LCI describes the limits that it will use to control loose surface contamination. However, the description is confusing and it is not clear whether 100 percent of the loose surface contamination limit will be used or 25 percent of the limit will be used as an action level for cleaning an area.

Issue 2

It appears that LCI is not correctly applying the methods of assessing surface contamination per Regulatory Guide 8.30, Regulatory Guide 1.86, and Fuel Cycle Policy and Guidance Directive 80.23. Industry is including radium (and potentially thorium-230) in the gross alpha measurements and this would appear to allow for higher release limits than would otherwise be allowed if radium (and potentially thorium) was excluded from the gross alpha measurements and viewed separately.

NRC staff noted that Regulatory Guide 8.30, Regulatory Guide 1.86, and Fuel Cycle Memorandum 83-23 are ambiguous with respect to surface contamination guidelines. There was an attempt by the NRC in the early to mid-90s to define the terminology (e.g., the meaning of "associated decay products") and application of surface release criteria. As an example, for contamination surveys, LCI proposes including radium (and potentially thorium-230) with overall natural uranium, but that is not the way the NRC has applied these limits in the past for industries other than uranium recovery. DPM value in table only applied to thorium and protactinium isotopes in secular equilibrium with uranium, not radium. NRC staff examples include dose calculations in SECY 98-155 and release criteria for the Molycorp York, PA facility.

Lost Creek Response (23 APR 2009):

Issue 1

LCI will clear up the language regarding loose surface contamination limits. LCI will specify that total contamination surveys will be done with survey meters. Removable contamination surveys will be performed with smears.

Issue 2

Going forward LCI stated that it will define surface contamination limits for natural uranium as noted by NRC staff so that radium (and potentially thorium-230) is not included in the release limits for natural uranium. It will use the same general strategy as for airborne contamination. First, LCI will characterize its operations with respect to potential contributors to surface contamination and make assumptions regarding the type and quantities of radionuclides being released. It will then confirm those quantities once operations have started with isotopic analyses. The result may be that LCI cannot release any equipment until specific isotopes and their associated release limits are evaluated.

Action:

LCI will submit a program consistent with its responses above.

August 5, 2009 Response:

Issue 1

Re-wording of paragraph on page 5-40 of the Lost Creek Technical Report:

Areas of the Plant where work with uranium is not performed and contamination is not expected (areas external to controlled process areas, e.g. offices, break rooms, etc) will be surveyed (spot checked) weekly for removable contamination (smear surveys). The ALARA contamination goal for these areas is background. Areas that are found to be contaminated with loose radioactivity will be cleaned immediately and re-surveyed. Total contamination instrument surveys will also be performed. If the total fixed contamination level exceeds the site contamination limit for non process, uncontrolled areas, these areas will be cleaned and re-surveyed until contamination is below this limit. If this cannot be achieved via simple cleaning methods (detergents, abrasive action, etc) the contaminated area may need to be extracted and replaced.

Site general contamination limits will be consistent with industry standards for release limits to unrestricted areas and are discussed in detail in response to issue 2 below.

Issue 2

In response to this issue, it is important and fundamental to recognize the radiological environment of a modern ISR as related to potential radionuclides of concern for which contamination surveys must be performed and unrestricted release limits established. As discussed in response to RAIs: 5.7.4(4), (item 3, *worker dose calculations*) studies performed in the late 1970s and early 1980s of radionuclide mobilization from several ISRs (see references provided in that response) and subsequent measurements at operating ISRs indicate a relatively small portion of the uranium daughter products in the ore body are actually mobilized by the lixiviant. The vast majority of secular equilibrium radionuclides remain in the host formation. Thorium 230 appeared to equilibrate and very little was actually removed by the process. The majority of the mobilized radium 226 (80—90 percent) which was estimated to be 5~15 percent of the calculated equilibrium radium in the host formation, followed the calcium chemistry in the process and resulted in radium carbonates / sulfates in the calcite byproduct waste streams. Little, if any Lead 210 was mobilized as the lead carbonate complexes formed in situ are virtually insoluble in the lixiviant processes studied. In addition to the fact that very little thorium is mobilized in situ, the ion exchange (IX) resin used in ISR facilities is specific for removal of uranium. Thorium compounds are not removed by the IX resin and are therefore not present in the process downstream of the IX columns (e.g., elution, precipitation, and drying circuits).

Accordingly, the existing, approved NRC guidance for unrestricted release of equipment / clearance limits for "Unat, U-235, U-238 and associated decay products" are applicable and appropriate for ISR plants. This includes the applicability of NRC Regulatory Guide 8.30, *Health Physics Surveys in Uranium Recovery Facilities*, 2002 (RG 8.30). Section B, *Discussion* indicates, " The contents of this guide conform with NRC's current licensing practice". We are unaware of any revisions of RG 8.30, subsequently issued NRC regulatory guides and/or NRC rules and regulations that supersede the continued use of RG 8.30 as issued in 2002.⁽¹⁾

Recommended surface contamination limits are defined in RG 8.30 in its Table 2 entitled "Surface Contamination Levels for Uranium and Daughters on Equipment to be Released for Unrestricted Use, on Clothing and on Non Operating Areas of UR Facilities." A footnote to RG 8.30 Table 2 indicates the stated contamination levels are taken from Regulatory Guide 1.86, "Termination of Operating Licenses for Nuclear Reactors" and from "Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct Source or Special Nuclear Material," August 1987. Both of these documents use identical radionuclide categories and quantitative limits although the 1987 document also specifies dose rate guidance (mrad/hr for beta gamma emitters)⁽²⁾. Since the title of RG 8.30 Table 2 indicates applicability of the table's values to uranium *and its daughters* (emphasis added), it is reasonable to assume that it was clearly intended to be applied to uranium recovery facilities with expected varying degrees of equilibrium and ratios of natural uranium series radionuclides.

Additionally, regarding contamination of skin and clothing, we again propose use of the RG 8.30 recommendations. In Section 4.7, *Contamination of Skin and Clothing*, it is specified "If alpha contamination of the skin or clothing of workers leaving a UR facility is found to exceed 1000 dpm/100 cm², an investigation of the cause of the contamination should be made and corrective action taken, if appropriate" However, for contamination of persons, background will be the ALARA objective.

The above is consistent with the historical application and standards of practice in the use of the radionuclide categories and associated contamination limits of these two documents (and by reference in RG 8.30) for releases of equipment to unrestricted areas, not only regarding USNRC / Agreement State licensees but also at US DOE and USACE TENORM contaminated sites. In addition to RG 8.30, see e.g.:

Regulatory Guide 8.21, *Health Physics Surveys For Byproduct Material At NRC - Licensed Processing And Manufacturing Plants* – Table 2, Footnote b – "Contamination limits for unrestricted (non-contamination-controlled) areas in this table are considered to be compatible in level of safety with those for release of facilities and equipment for unrestricted use as given in Regulatory Guide 1.86..... "

Regulatory Guide 8.23, *Health Physics Surveys at Medical Institutions* – Table 3 is the same as RG 1.86, Table 1

US DOE Order 5400.5, *Radiation Protection of the Public and Environment* – Figure IV – 1 is the same as Table 1, RG 1.86

US Army Corps of Engineers EM 385-1-80, *Radiation Protection Manual*, 1997- Table 6-4 is essentially identical to Table 1, Regulatory Guide 1.86

Additionally, for ISR license applicants, NUREG 1569, *Standard Review Plan for In Situ Leach Uranium Extraction License Applications* provides the current NRC guidance. We are unaware of any revisions of NUREG 1569 and/or subsequently issued NRC regulatory guides that supersede this document. It states that the applicant must ensure that "appropriate criteria are

established to relinquish possession or control of equipment or scrap having surfaces contaminated with material in excess of the limits specified in Table 5.7.6.3" which is taken from Table 1 of Regulatory Guide 1.86. (See page 5-31 of NUREG 1569). Furthermore, NUREG 1569 states (page 5-30):

"The contamination control program is acceptable if it meets the following criteria:

Radiation surveys of workers will be conducted to prevent contaminated employees from entering clean areas or from leaving the site in conformance with guidance in Regulatory Guide 8.30.....

The proposed contamination control program is consistent with the guidance on conducting surveys for contamination of skin and personal clothing provided in Regulatory Guide 8.30....

Action levels for surface contamination are set in accordance with Regulatory Guide 8.30, Section 4.

Regarding volumetric contamination limits, it is important to note that NRC is in the process of consolidating decommissioning guidance ⁽¹⁾, and has not yet issued final rules on volumetric contamination limits and the guidance in RG 8.30 only addresses surface contamination. A potentially useful reference for addressing volumetric contamination until such time as NRC issues its final rules is *American National Standard – Surface and Volume Radioactivity Standards for Clearance*, ANSI/HPS N13.12-1999. This national standard, approved by the American Standards Institute and published by the Health Physics Society represents the best consensus science currently available. The primary criterion of this standard is to provide for public health and safety to an average member of a critical group such that the dose shall be limited to 10 $\mu\text{Sv}/\text{y}$ (1.0 mrem/y) Total Effective Dose Equivalent (TEDE), above background, for clearance of materials from regulatory control. Additionally, since the recommended surface contamination limits in ANSI 13.12 are based on 1 mrem/year and are consistent with the RG 8.30 criteria, this provides a dose/risk framework supportive of the appropriateness of the RG 8.30 guidance. ⁽³⁾

NRC cited SECY 98-155 as containing examples of dose calculations. Attachment 6 to SECY 98-155 appears to be the portion of interest. However, according to the NRC Public Document Room staff, Attachment 6 is unavailable to the public due to national security concerns.

Conclusion

To the best of our knowledge and belief, the personnel contamination control guidance and surface contamination criteria for release of equipment and material to unrestricted areas as defined in RG 8.30 represents the current, approved NRC staff position. NUREG 1569 similarly represents the currently approved guidance to NRC staff against which an ISR applicant's source material license submittal is to be reviewed.

Studies performed over many years at operating ISRs indicate very little of the uranium progeny in the host formation is mobilized by the lixiviant and the selective nature (for uranium) of the

resins used results in very small amounts of uranium progeny being moved forward into the elution, precipitation and back end processes of modern ISRs. Historical application by multiple Federal agencies is clear that the category of "Unat, U-235, U-238 and associated decay products" (as originally used in Regulatory Guide 1.86 and "Guidelines for Decontamination...for Byproduct, Source or Special Nuclear Material"), incorporated by reference into RG 8.30, is appropriate for the radiological environment of ISRs. Analysis performed to assess the dosimetric / risk based consequences of the application of these limits indicate they are protective and provide an appropriate standard of care. Accordingly, the applicable recommendations and guidance provided in RG 8.30 and NUREG 1569 will be incorporated into the contamination assessment and control elements of the Lost Creek ISR radiation protection program.

⁽¹⁾ NUREG 1757, *Consolidated Decommissioning Guidance - Decommissioning Process for Materials Licensees* (2006) has been recently issued. The objectives of the consolidation effort were to consolidate existing guidance into a single (three-volume) document, to update the guidance as needed, and to make the guidance more risk-informed and performance-based. More than 80 documents were evaluated in developing the consolidated guidance. NUREG-1757 updates and builds upon the risk-informed approach used in the NMSS Decommissioning Handbook, NUREG/BR-0241, *NMSS Handbook for Decommissioning Fuel Cycle and Materials Facilities*, March 1997 and the *NMSS Decommissioning Standard Review Plan*, NUREG-1727, September 2000. The three volumes of NUREG-1757 supersede these documents in their entirety.

NUREG 1757 takes a risk-based approach to establishing radiological release criteria. The fundamental risk based criteria of NUREG 1757 is that residual radioactivity, distinguishable from background, does not result in a calculated dose from all pathways to the average member of the critical group in excess of 0.25 mSv/y (25 mrem/y). Accordingly, generic surface contamination and/or volumetric limits (Bq/100cm² or Bq/gram) are not specified but may be determined on a site-specific basis to ensure compliance with the fundamental risk / dose guidance. It should be noted that NUREG 1569, *Standard Review Plan for In Situ Leach Uranium Extraction License Applications*, references NUREG 1775, Revision 1, *Multi-Agency Radiation Survey and Site Investigation Manual, USNRC 2000 (MARSSIM)* for acceptable survey methods for decommissioning ISL facilities. MARSSIM provides the technical basis for many of the methods described in NUREG 1757.

⁽²⁾ It is also of interest to note that Policy and Guidance Directive FC 83-23: Termination of Byproduct Source and Special Nuclear Material (1983), also referenced by NRC staff in their letter to LCI of April 23, 2009, uses the 1982 version of "Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct Source or Special Nuclear Material," as its ENCLOSURE 2 with the identical radionuclide categories and contamination limits as the 1987 version as well as with RG 1.86.

⁽³⁾ ANSI 13.12, in its Table1 provides surface and volumetric contamination limits associated with an annual dose to a member of the public of 1 mrem/yr. Of particular relevance to a uranium recovery facility, the surface contamination limit recommended for uranium in association with its decay chain is 6,000 d/m/100cm². This supports the historical standards of

practice that the recommended limits of RG 8.30 for uranium and its daughter products (1,000 removable, 5,000 total avg.) are protective and represent an appropriate standard of care since ANSI 13.12 uses a 1 mrem / yr public exposure criteria as a dose/ risk basis for its recommended contamination limits.

5) Preoperational and operational environmental sampling

RAIs: 2.9(3) & (4), 5.7.7(2)

Discussion (23 APR 2009):

Preoperational sampling:

1) LCI does not address radon daughter deposition on vegetation as indicated in its application and this RAI question 2.9(3). The vegetation sampling data submitted was intended to address a future yellowcake dryer only. These locations would not necessarily be the same as areas maximally impacted by radon daughter deposition. As an example, NRC staff discussed the lack of sampling at the general locations near SEB1 and SEB2, which are the highest dose locations according to the modeling performed by LCI.

2) LCI does not present a coherent air sampling program:

- Selection of air particulate locations does not appear to comport with criteria in Regulatory Guide 4.14 (e.g., para 1.1.1 of this regulatory guide regarding prevailing wind direction, location of estimated maximum concentrations of radioactive materials, etc.), nor is there enough discussion to determine why it chose the given locations.
- There are no co-located particulate samplers with radon sampling stations as recommended in RG 4.14 nor is there a discussion on why this is the case.

Operational Environmental monitoring:

3) LCI does not address Pb-210 in its proposal to not monitor for airborne particulates.

4) NRC staff commented that the evaluation for vegetation, food, and fish was a good analysis but asked if LCI had analyzed Note (O) in Regulatory Guide 4.14 regarding operational vegetation or forage sampling for its site.

5) LCI does not address soil sampling, as recommended in RG 4.14 for an operational radiological monitoring program, and as asked RAI question 5.7.7(2). As pointed out in the question, and identified in the application, radon daughters will accumulate in soil, among other places. RG 4.14 includes Pb-210, a radon daughter, as a radionuclide to sample.

Lost Creek Response (23 APR 2009):

Preoperational sampling:

LCI understood NRC staff concerns and will reevaluate its response.

Operational Environmental monitoring:

LCI understood NRC staff concerns and will reevaluate its response.

Action:

LCI stated that it will address these issues.

August 5, 2009 Response:

Preoperational sampling:

1) *Vegetation Sampling*

LC ISR, LLC has begun collecting vegetation samples to address this concern, and results will be presented to NRC when the sampling and analysis are complete. Seven locations were selected for additional vegetation sampling (See Attachment 1):

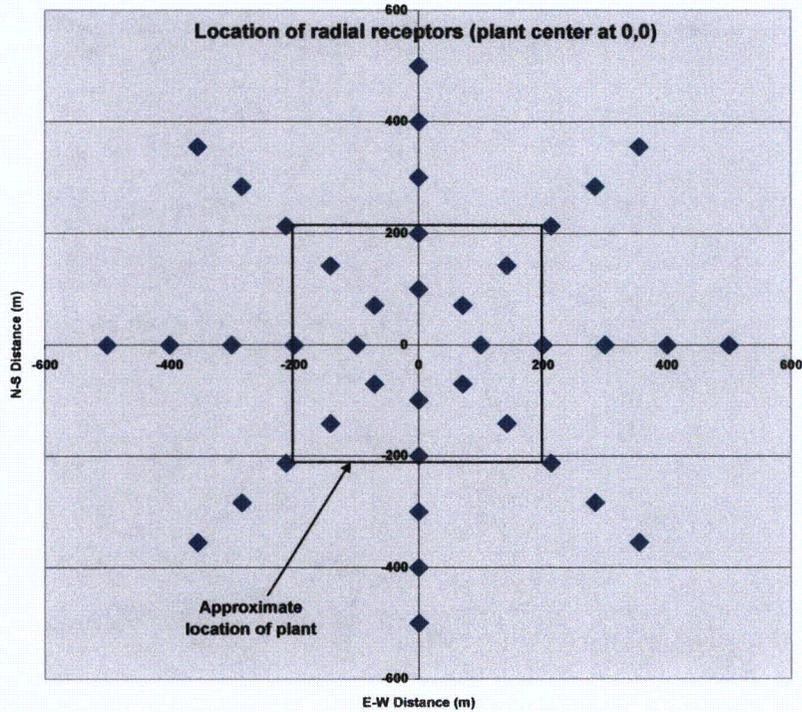
- Two locations (D & E) where total ground concentrations were predicted to be the greatest during operations, based on the MILDOS modeling described below;
- Four locations (F,G,H,I) where the baseline direct gamma scan survey indicated elevated gamma activity;
- One location (J) where the baseline direct gamma scan survey indicated comparatively low gamma activity, that is upwind of the Plant and where Project-related radon deposition is expected to be low or non-existent.

The first of three sets of samples were collected on June 24-25, 2009, and two more sets of samples will be collected at approximately two-week intervals. Samples will be analyzed for natural uranium, radium-226, thorium-230, lead-210, and polonium-210.

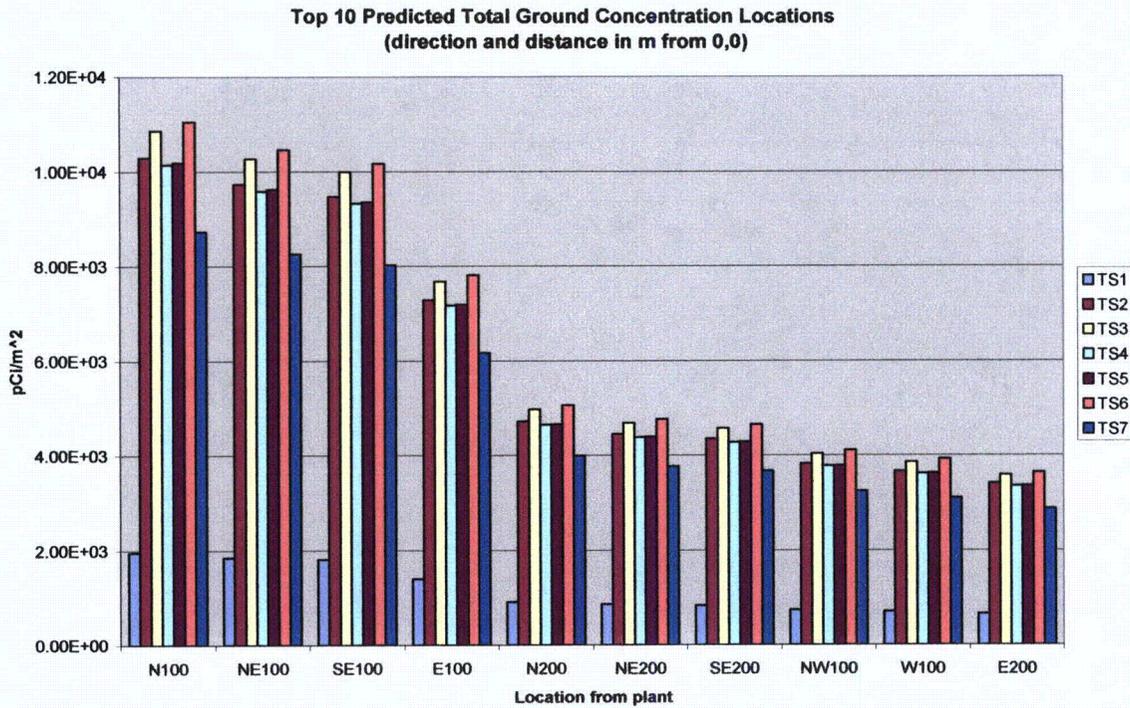
MILDOS modeling was conducted to determine the locations of maximum ground concentrations of radon surrounding the permit area and plant site. No operational parameters were changed from modeling that was done for the license application. Since only ^{222}Rn is released from the facility, the ground concentrations represent radon decay products. In modeling done for the license application a set of 17 boundary receptors was modeled, all of which were at the boundary of the permit area.

A grid with a series of 37 receptors was devised, spaced surrounding the plant site and permit area out to distances of over 2 km. Results for the grid receptors showed maximum concentrations closer to the plant site than anticipated. One difference between the modeled boundary receptors and the grid receptors is that the grid receptor elevations were set to zero in all cases. After analyzing the grid receptor results, it was decided to create a near field set of receptors in a radial pattern surrounding the plant location as discussed below.

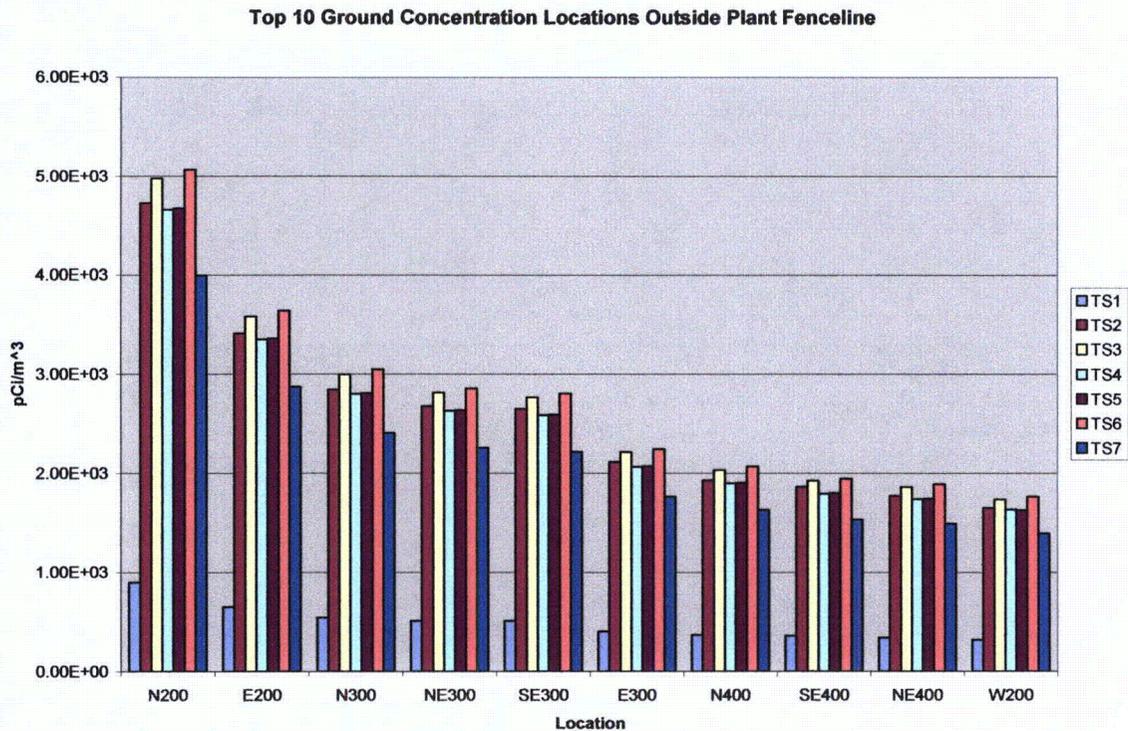
A series of 40 receptors was created in each of 8 compass directions in 100 m increments out to 500 m as shown below. The release point for the plant is point 0,0. Receptor locations were given names with the direction and distance from the origin. Receptor N100 is north of the plant center 100 m. Likewise, SE400 is 400 m southeast of the plant center. The approximate location of the plant fence line is shown for reference.



Results of the 40 radial receptors were compiled using total ground concentrations for the current time step (1 yr). Total concentrations include concentrations of Po-218, Pb-214, Bi-214 and Pb-210. Results of the top ten receptor locations for each time step are shown below.



The four highest (and 8 of the top 10) predicted ground concentrations in each time step are within the approximate plant boundary. The remaining two, N200 and E200 are approximately on the plant fence-line. Removing the locations that are closer than 200 m yields the following results.



The maximum value of 5,000 pCi/m² is shown at location N200, which is 200 m straight north of the plant center. Assuming that all activity resides in the top 1 cm of soil which has a density of 1.2 g/cm³, this equates to approximately 0.4 pCi/g of soil. Over the course of a year, it is unlikely that such low concentrations would be detected in soil or vegetation sample.

Isopleths of ground concentration for the radial receptors are shown below for year 6, which has the greatest releases according to model output.

2) Air Sampling

Selection of Air Particulate Sampling Locations. The baseline radiological air particulate sampling is described in a Technical Memorandum submitted to the NRC in January 2009 in response to RAI's on the Lost Creek Technical Report. The five sampling locations (Attachment 2) were selected in November 2007 based on: the Regulatory Guide 4.14 requirements; site knowledge; and available meteorological data from the on-site meteorological station and the Lost Soldier and Muddy Gap stations, which are about 12 and 28 miles northeast of the Lost Creek Permit Area, respectively (TR Section 2.5).

Site HV-2 was selected to represent the area where airborne radionuclide concentrations related to plant operations were predicted to be the highest, and is located immediately downwind of the 10-acre Plant Site. Site HV-4 was placed at the eastern Permit Area boundary, generally downwind of the Plant Site. Site HV-5 was located at a Permit Area boundary, less than one mile northwest of the Plant Site. Site HV-3 was selected to represent background conditions, since it is the location furthest from the Plant Site in a westerly, generally upwind, direction. Site HV-1 was located at the closest residence, 17 miles northeast of the Permit Area in Bairoil, WY.

Based on the MILDOS modeling previously submitted (with receptors at the Permit Area boundaries), and the more recent MILDOS modeling described above (with receptors extending radially from the Plant), the air particulate sampling locations represent the range of airborne radionuclide concentrations that are predicted at the site. As an example, air particulate concentrations at Site HV-3 are unlikely to be affected by Project activities, therefore, as anticipated, this location represents background conditions.

Co-Located Samples. Radon and gamma passive samplers were installed beginning in November 2006 (Attachment 3). At that time, the Plant Site had not been determined. Sampling locations were selected to represent: the center of the Permit Area (URPA-9); the upwind (URPA-7) and downwind (URPA-10) locations (based on site knowledge and available meteorological data); the southern Project Area boundary (URPA-8), which is closest to the Sweetwater Mill; and the closest residence (URPA-1). In February 2007, an additional sampling location (URPA-13) was added in response to the expansion of the proposed Permit Area.

The URPA-10 radon and gamma passive sampler site is co-located with the HV-4 air particulate sampling site, and the URPA-1 site is co-located with the HV-1 site. When reasonable estimates for the Plant Site location and predicted elevated radionuclide concentrations became available, it was determined that the best course of action was to select air particulate sampling locations independent of the passive samplers, in most cases, to provide more coverage of the site but still have some overlap.

Operational Environmental monitoring:

There will be no significant release of airborne particulate radionuclides during production since the LCI facility will not have the types of operations that have the potential to produce dust effluent (i.e., ore crushing and grinding). However, the decay of radon attributable to the facility may result in deposition of a very small amount of radon decay products, including short-lived Po-214, on surface soils. Polonium-214 will decay rapidly to Pb-210. The MILDOS analysis indicates that the maximum approximate surface contamination of 5,000 pCi/m² at a point 200

m north of the plant center. The estimated increase in the average Pb-210 activity concentration in the top 5 cm (2 inches) of soil would be approximately 0.06 pCi/g, assuming a bulk soil density at that depth of 1.6 g/cm³. This incremental concentration would be indistinguishable from background. Background Pb-210 concentrations range from 0.4 to 4.9 pCi/g (TR Table 2.9-1).

In agreement with RG 4.14, LCI will monitor radon gas and direct radiation, at the same locations as in the preoperational monitoring (Attachment 3) plus at those additional operational locations described in LCI TR Section 5.7.7 (Attachment 4). Additionally, radon gas and direct radiation measurement will also be co-located with the background HV-3 particulate air monitor, and monitors HV-2 and HV-5 (Attachment 2). HV-2 is currently located within the location of the plant, so practically it must be relocated slightly and will be repositioned just outside the fence line nearest the MILDOS maximum point discussed above. There will then be a total of eleven locations with radon gas and direct radiation measurement; five of which will be co-located with the particulate air monitors.

With regard to RG4.14, Footnote O to Tables 1 and 2, operational vegetation samples will not be taken since according to BLM guidance it takes approximately 144 acres annually at this location to support one head of livestock. Such sparseness of forage cannot conceivably result in significant cattle exposure through this pathway. Additionally, there are no cattle on these acreages used for human consumable milk production, so that the milk consumption pathway is not a consideration. The well pattern area as well as the plant will be fenced off from cattle, so cattle exposure through spillage is also of negligible concern.

Soils will be monitored annually in agreement with RG4.14 at each of the five air particulate locations.

6) Dose to public/Effluent monitoring

RAIs: 5.7.1(1), 5.7.7(1)

Discussion (23 APR 2009):

NRC staff stated that it was not clear that LCI went through the process of identifying the most exposed member of the public and included onsite areas as is required by 10 CFR 20.1302. Furthermore, the staff notes that §40.65 requires quantification of radionuclides released to unrestricted areas. This can either be calculations or monitoring; however, calculations or modeling must be confirmed by sampling.

The staff also noted that LCI should be careful to state that certain members of the public would not exceed 2mrem/hr. This may not necessarily be true, and could be grounds for a citation if an inspection indicated that this was false.

LCI should refer to 60 FR 36038 for a revised definition of "member of the public" and "public dose."

Lost Creek Response (23 APR 2009):

LCI responded that it will review its data and identify the most affected member of the public, and it will provide more information regarding effluent monitoring/sampling. Specifically, it will identify the equipment evaluated and its analysis for continuous sampling and look at grab sampling. NRC staff asked if LCI had evaluated EPA testing method 114 for radon, as previously suggested. LCI noted that it did not review this issue in more detail pending NRC staff's review of this issue.

When LCI questioned where the correct place was to monitor for compliance with 10 CFR 40.65 as it pertained to controlled radon discharges (as opposed to uncontrolled, or "out the door" radon discharges), NRC staff replied that it will generally be at the stack discharge. There was some confusion with LCI as to what constituted a stack and NRC staff replied that in this scenario it would include piping, etc., that directed airborne effluent outside of the facility into the unrestricted area. NRC staff also suggested that LCI review guidance in NUREG-1736 for compliance with 10 CFR 20.1302 in regard to measurement locations for airborne effluents at boundaries of unrestricted areas.

Action:

LCI will reevaluate its response and submit information as discussed above.

August 5, 2009 Response:

Health physics staff will regularly evaluate and document gamma exposure rates at the facility during operations using properly selected and calibrated instruments. Site visitors will not be exposed to radiation in excess of 2 mrem in one hour.

LCI has examined current options available to allow stack monitoring for intermittent releases of radon gas during IX column purge events. These planned stack releases potentially involve approximately a curie of Rn-222 over 30 minute periods, daily (The annual release from these emissions will be approximately equal to the annual radon release from any typical 3-4 square miles of the planet's surface.). Discussions with the manufacturers of instruments potentially capable of radon continuous stack monitoring indicate that significant modification of instruments such as the Pylon CRM-1® would be necessary to allow the instrument to function reliably at the planned release concentrations in a high-velocity flow stack environment. Per the

CRM-1 manufacturer, "The radon detector is comprised of a Lucas type cell. The gas is drawn from a remote location by a rotary vane pump into the Lucas-type cell. The flow is then diverted through a filter to remove the radon daughters before entering the Lucas-type cell and returned to the exhaust vent. As the radon decays it emits an alpha particle that strikes the silver activated zinc sulfide coating of the cell. The energy of the alpha particle is converted to a light pulse by the phosphor. The light pulse is amplified by the PMT and counted by the CRM-1 over the user programmable measurement interval." The manufacturer's representative indicated that, given some experimentation, greatly reducing the active area of the CRM-1 flow-through Lucas chamber could be developed into an acceptable approach to providing such a stack monitor. However, given the need for development work, LCI concludes that such a possible system is not yet an actual device acceptable for license reference. Grab sample evaluation of radon concentrations during a planned stack release, utilizing an activated charcoal system followed by gamma analysis to estimate stack radon concentration, could be utilized, but would not provide a continuous readout of stack concentration over time. Placement of passive, integrating in-stack monitors, such as Track Etch® or similar systems, is also a possibility, but planned stack concentrations during release events are high enough that significant development work would be required to ensure accurate radon concentration measurements using such devices.

More significantly, however, while planned radon stack releases may involve a significant fraction of the facility's and the active site's total radon releases, it is clear that other significant potential release sources, including well fields under development, other facility ventilation locations, and facility entrance/exit doors, will contribute an additional component to the overall facility source term (For discussion of relative magnitudes see: Brown, S and Smith, R, 1980, *A Model for Developing the Radon Loss (Source) Term for a Commercial In Situ Leach Uranium Facility*, In: M Gomez (Editor), *Radiation Hazards in Uranium Mining – Control, Measurement and Medical Aspects*, Soc. Min. Eng., pp 794-800.)

Given this situation, LCI therefore proposes the placement of Track Etch® integrating radon monitoring devices onsite and offsite, at locations with the potential for radon concentration maxima based on site-specific modeling (see LCI's responses to RAI Item 5, "Preoperational and operational environmental sampling" in this document.). Such monitors would be in addition to Track Etch® monitoring as previously specified in the LCI license application, and would be replaced and analyzed at frequencies appropriate to anticipated concentrations at each location. Given that radon in-stack monitoring, even if eventually developed into well-tested technology, is capable of quantifying only a portion of total site radon releases, integrated, continuous monitoring of actual radon concentrations at locations of interest, including the locations of modeled, site-specific maximum concentrations (using Lost Creek facility area-specific meteorological data), will provide the most reliable estimate of exposure associated with overall site radon releases. MILDOS will be applied to the radon measurement results to determine the quantity of radon released to unrestricted areas.

7) Spill cleanup criteria

RAIs: 6.5(2), 7.4(1)

Discussion (23 APR 2009):

NRC staff stated that LCI needs to develop soil cleanup criteria for uranium. Currently, LCI intends to use the background radiological survey to develop cleanup criteria; however, this is for radium only.

LCI has made commitments throughout its application to clean up spills that exceed 10 CFR 40, Appendix A, Criterion 6, levels (see, for example, 5.7.1(3)(b) and (d), 5.7.7(2) and 7.4(1)). LCI cannot comply with its commitments unless it has established acceptable levels beforehand.

Lost Creek Response (23 APR 2009):

LCI stated that developing uranium soil cleanup criteria is premature since the surrounding land use may change.

NRC staff responded that it can make conservative assumptions regarding land use for uranium cleanup criteria using an appropriate computational methodology such as RESRAD, for example. Staff can then confirm those assumptions prior to decommissioning. LCI agreed to reevaluate its response in accordance with this discussion.

Action:

LCI will address this issue and submit a plan that incorporates its response as indicated above.

August 5, 2009 Response:

Accident scenarios for ISR facilities are described in detail in NUREG/CR-6733 (NRC, 2001). Potential doses from such incidents were estimated based on the assumption that a spill would not be cleaned up immediately and would be allowed to dry (see response to item (1) U₃O₈ / Accident scenarios, RAIs: 4.1(5), which provides a comparison of assumptions in NUREG/CR-6733 vs LCI's circumstances). In such a case, the most significant potential route of exposure to workers and members of the public would be limited to inhalation of airborne radioactive material. However, with regard to residual contamination remaining after spill cleanup is completed, the doses to workers would include direct radiation dose as well as inhalation of particulates. The dose to a member of the public with unrestricted access to and use of the impacted area could include a variety of pathways.

LCI will conduct operations, to include spill cleanup, in agreement with the ALARA principle and the "member of the public" and worker dose requirements of 10 CFR 20 and the "member of the public" requirements of 40 CFR 190.10. However, since access to spill locations will be restricted during the years of operation, only those exposure pathways consistent with the site access restrictions and existing land use will be used to meet these regulatory requirements.

LCI's direct radiation surveys and correlations to measured soil Ra226 ($R^2=0.88$; TR Figure 2.9-7) and soil Ra226 correlations to measured soil uranium ($R^2=0.73$; TR Figure 2.9-8) provide the basis for uranium and radium soil background concentrations characterized over the LCI site. However, whenever spills occur, LCI will collect additional soil samples outside the spill margins to further characterize the soil radionuclide concentrations so that when combined with the

radionuclide analysis of the spill content, accurate cleanup levels can be established to meet the decommissioning "Radium Benchmark Dose" of 10 CFR 40, Appendix A, Criterion 6. This will assure there will be no appreciable radionuclide migration off the spill location, and final decommissioning will be facilitated.

The following discussion illustrates how LCI will use RESRAD analysis to establish soil cleanup criteria, and presents proposed initial cleanup criteria.

Activity Concentration Clean-up Criteria for a Spill at the Lost Creek Facility

NUREG 6733 (NRC, 2001) describes spill scenarios involving solid or liquid materials: thickener failure and spill, pregnant lixiviant spill, and loaded resin spill. The consequences of such accidents are described in our response to item (1) U3O8 / Accident scenarios, RAIs: 4.1(5).

NUREG 6733 assumes no initial cleanup in its risk assessments for the spill scenarios. LCI is committed to taking all necessary precautions to ensure that such spills do not occur. However, in the unlikely event of a spill of solids or liquids containing radioactive material, appropriate actions will be taken initially to remove spilled material and clean up the impacted areas to levels such that residual radiation doses to workers from the spill following initial cleanup would be less than 100 mrem per year (LCI perceives this as an initial ALARA target for workers) and doses to members of the public, no greater than 25 mrem per year. Spill-impacted areas will be cleaned up to reduce doses as low as reasonably achievable (ALARA) below these levels. Further cleanup of impacted areas, if necessary to meet criteria for unrestricted use, would be included in the final decontamination and decommissioning of the facility. LCI will use RESRAD as appropriate, using analysis results from cleanup samples to verify that the above goals have been met.

The following analyses assume that a spill inside the restricted area would impact workers during operations, and that a spill outside the restricted area could impact members of the public with unrestricted access to, and use of, the impacted area. Since the intent of this analysis is to develop criteria for residual contamination after spill cleanup, it does not need to address dose from the spill itself or resulting cleanup operations to workers (whose dose will be controlled under the in place, approved radiation protection and ALARA programs) or members of the public (who cannot have unrestricted access during licensed operations or who would not have access to affected unrestricted areas during cleanup).

Thickener Failure and Spill

NUREG 6733 postulates a spill of 73,500 gallons of slurry containing 24,200 kg of U₃O₈. Assuming such an incident was to occur at LCI, the cleanup criteria would be dependent on the potential dose from natural uranium. According to NUREG 6733, the sole substantial radiological hazard would be inhalation of airborne particulate matter. However, based on a RESRAD analysis, the primary contributor to dose from natural uranium would be external exposure (ground) presumably from beta and gamma radiation from the immediate decay products of U-238 (Th-234 and Pa-234m).

Any portion of the spill inside a building or containment would be cleaned up immediately and would not have the opportunity to dry out and become airborne. Therefore, doses to workers would be limited to the initial cleanup phase. The criterion for immediate cleanup within a building or containment would be based on the presence of visible residues, i.e., any visible loose spill material would be removed. Liquids that are absorbed into surface material such as concrete would not present a significant inhalation hazard as the uranium would not become airborne. A spill outside the building with the potential to contaminate soils would also be cleaned up immediately before the material could dry sufficiently to become an airborne dust hazard.

Spills within the restricted area will be cleaned up to levels that are ALARA. At a minimum, the impacted areas will be cleaned up to levels that would limit the residual, post cleanup dose to a worker to less than 100 mrem per year. Based on a RESRAD analysis, a U-nat concentration in soil equal to 100 pCi/g would result in an annual dose of 2.5 mrem/year. Since the dose is proportional to the concentration of uranium in soil, a cleanup level of 4000 pCi/g would result in an annual dose to a worker spending all of his or her 2,000 hr working year in the spill area approximately equal to 100 mrem per year, 76% from direct external exposure, 14% from inhalation of particulate matter, and 10% from soil ingestion. The RESRAD default dust concentration (0.1 mg/m^3) was used in the analysis. However, an increase in the assumed dust concentration to 1 mg/m^3 (to allow for possible LC site wind and dust conditions) would decrease the estimated cleanup criterion to 1,800 pCi/g due to the increased dose from inhalation of airborne particulate matter. An appropriate cleanup standard for spills within the restricted area, based on protection of workers, including a reasonable safety factor, is 1,500 pCi/g U-nat. LCI proposes this as the initial cleanup criterion for a spill of this nature.

Based on a RESRAD analysis, the dose to a member of the public at 100 pCi/g U-nat would be approximately 8 mrem per year, approximately 60% from direct external exposure, 10% from inhalation of particulate matter, and 30% from ingestion of locally grown plants, meat, and milk, as well as ingestion of soil. The estimated U-nat concentration in soil resulting in a dose of 25 mrem per year would be approximately 300 pCi/g above soil background concentration. This analysis is very conservative because it includes food chain pathways even though it is unlikely that food would be raised in the impacted area. LCI proposes this as the initial cleanup criterion for a spill of this nature in an unrestricted area.

Pregnant Lixiviant and Loaded Resin Spills

In its risk analysis, NUREG 6733 assumes the pregnant lixiviant and loaded resin contains Ra-226 at a concentration of $3.4\text{E}3$ pCi/L and U-nat at a concentration of $1.7\text{E}5$ pCi/L. The short-lived decay products of Rn-222 were assumed to be in equilibrium with the Ra-226. As with the thickener spill scenario, the impacted area would be cleaned up immediately. The criteria for cleanup were calculated assuming a maximum annual dose to a worker of 100 mrem per year and 25 mrem per year for a member of the public. The RESRAD analysis was performed assuming a nominal U-nat concentration of 100 pCi/g in soil and a Ra-226 concentration of 2 pCi/g in soil, the ratio of the nuclides specified in NUREG 6733. The appropriate clean up criterion was determined by scaling the nominal concentration.

Based on the most conservative RESRAD analysis, assuming an air particulate concentration of 1 mg/m³, (again allowing for possible LC site wind and dust conditions) the estimated annual dose to a worker at a U-nat concentration in soil of 100 pCi/g and a Ra-226 concentration of 2 pCi/g, was approximately 10 mrem/year. Therefore, the cleanup criterion for U-nat would be 1,000 pCi/g with 20 pCi/g Ra-226 above soil background levels. LCI proposes this as the initial cleanup criterion for a spill of this nature.

The RESRAD-estimated dose to a member of the public from residual contamination after a spill of pregnant lixiviant (loaded resin spills will only occur in restricted areas since LCI does not intend to ship loaded resin at this time. If, in the future, resin is shipped from or to the site, additional analysis will be performed), assuming a U-nat concentration of 100 pCi/g and a Ra-226 concentration of 2 pCi/g, was 20 mrem/year. Therefore, based on a dose limit of 25 mrem per year, the cleanup criterion for members of the public would be 120 pCi/g U-nat, 2.2 pCi/g Ra-226. LCI proposes this as the initial ALARA target cleanup criterion for a spill of this nature; however, following a spill of this nature, LCI will use RESRAD with appropriate current land use and actual spill concentrations of Ra-226 and U-nat to re-determine the appropriate and justifiable cleanup criterion. Regardless, the cleanup criterion will meet the decommissioning "Radium Benchmark" of 10 CFR 40, Appendix A, Criterion 6.

Yellowcake Spill

LCI will apply the same cleanup criterion for a yellowcake spill, as for the thickener spill since in both cases the only nuclide of concern is uranium (NUREG 6733).

In all cases, LCI will clean spills up as soon as practicable and will restrict access to the impacted area until the cleanup criteria are met. The above calculations are based on the assumed concentration ratio of U-nat to Ra-226 in the plant radioactive materials. The criteria will be adjusted if site specific data show a different assumption should be used.

SUMMARY OF LCI PROPOSED INITIAL CLEANUP CRITERIA:

Exposure Scenario	Worker (Above background)	Public (Above background)
Thickener and Yellowcake	U-nat = 1500 pCi/g	U-nat = 300 pCi/g
Pregnant Lixiviant and Loaded Resin	U-nat= 1000 pCi/g Ra-226 = 20 pCi/g	U-nat = 120 pCi/g Ra-226 = 2.2 pCi/g

References:

U. S. Nuclear Regulatory Commission (NRC). 2001. A Baseline risk-Informed, Performance-Based Approach for In Situ Leach Uranium Extraction Licensees. NUREG/CR-6733. June.

RESRAD Version 6.3. Argonne National Laboratory.

8) Pre-reclamation radiation survey RAIs: 6.5(4)

Discussion (23 APR 2009):

NRC staff stated that the application has not adequately demonstrated that the gamma/Ra-226 correlation can be used to form a correlation with other radionuclides in order to meet 10 CFR 40, Appendix A, Criterion 6 criteria.

The application and literature references submitted by the LCI are focused on correlating gamma survey readings with surface soil Ra-226 concentrations. In fact, the only correlation between Ra-226 and uranium is found in this application in one figure (Fig. 2.9-8). This correlation has an R-squared value of 0.73. Further, there is no discussion of how this second order correlation (i.e, Uranium correlated to Ra-226, which itself is correlated to a gamma survey) and the associated errors will affect the ability to detect, for example, uranium.

Lost Creek Response (23 APR 2009):

LCI and its contractors will discuss this issue and propose a solution.

Action:

LCI will submit the results of its analysis discussed above to NRC staff.

August 5, 2009 Response:

LCI indicated in its response for comment 7 (c), Section 2.9 in the responses to NRC 11/6/08 comments for the Lost Creek Project (dated January 16, 2009), the following passages which are believed pertinent in helping to address this comment:

“Regulatory Guide 4.14 recommends that 40 surface soil samples be collected in a radial grid surrounding the mill, and 10% (four) of these samples be analyzed for uranium (NRC, 1980). In addition, it recommends that soil samples collected at the five air particulate monitoring stations be analyzed for uranium. Therefore Regulatory Guide 4.14 recommends that fewer than ten surface soil samples be analyzed for uranium.”

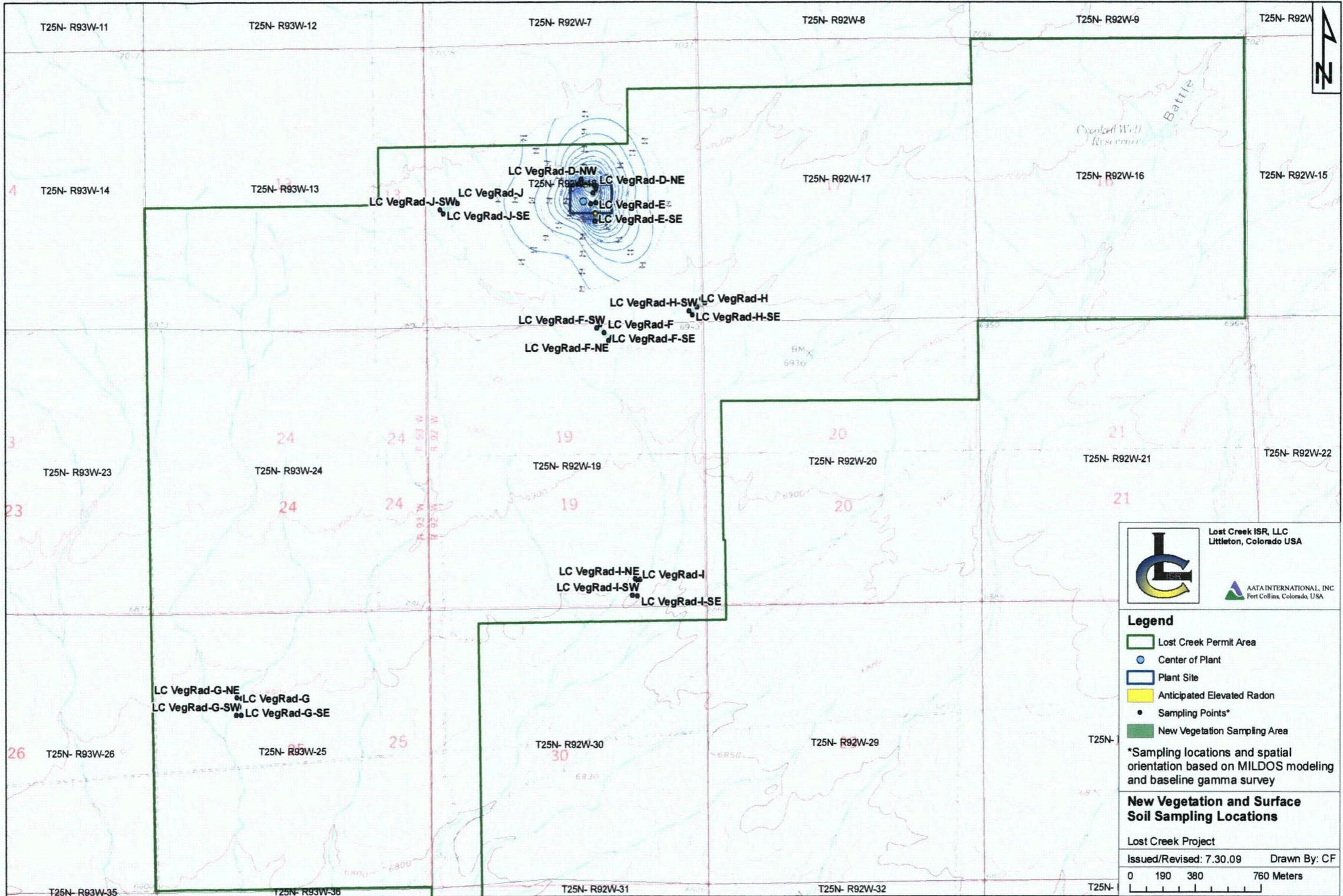
“At the Lost Creek site, ten surface soil samples were collected in a roughly radial pattern relative to the center of the site. These samples were analyzed for Ra-226, U-nat, Th-230, and Pb-210. For characterizing baseline uranium in surface soils, this sampling design is reasonably consistent with the Regulatory Guide 4.14 recommendations, and should satisfy the basic intent and technical basis of the regulatory guidance. Furthermore, the gamma survey goes far beyond Regulatory Guide 4.14 recommendations and this information can be used to indirectly estimate approximate baseline concentrations of both Ra-226 and uranium in surface soils anywhere on the site.”

“The intensive gamma survey performed across the entire site helps to overcome limitations of a Regulatory Guide 4.14 design for characterizing spatial variability in baseline concentrations of Ra-226 in surface soils. The statistical correlation between Ra-226 and uranium suggests that survey data can also be used to indirectly infer approximate uranium concentrations. Had the baseline soil sampling and gamma survey designs for this site strictly adhered to RG 4.14 guidelines, far less spatial information relevant to the assessment of potential uranium contamination due to spills and accidents would be available.”

LCI acknowledges that there is uncertainty in the second order correlation between gamma readings and uranium, via intermediate relationships for these parameters with soil Ra-226. Estimation error for each intermediate correlation relationship is additive and the total uncertainty of a direct correlation between gamma radiation and uranium in soil would include the combined uncertainties in gamma/Ra-226 and Ra-226/uranium correlations.

Although considerable uncertainty exists for estimating baseline uranium concentrations based on gamma readings, the basic recommendations and intent of the regulatory guidance for characterizing baseline uranium and other radionuclides in soil at Lost Creek are satisfied by the direct soil sampling and analysis that was conducted across the site. Additional uranium concentration estimates, based on the intensive gamma survey and the second order correlation, represent added data analysis (performed in addition to current regulatory guidance) in an attempt to improve knowledge of this parameter and reduce overall uncertainty in baseline characterization of uranium in soils at the site. These data will be helpful in the event of spills or other events during the operational life of the facility, to provide evidence of pre-existing conditions at the site in addition to the soil sampling data discussed above. See also the response to item 7 regarding additional soil sampling in the event of a spill.

ATTACHMENT 1




Lost Creek ISR, LLC
 Littleton, Colorado USA


AATA INTERNATIONAL, INC.
 Fort Collins, Colorado, USA

Legend

-  Lost Creek Permit Area
-  Center of Plant
-  Plant Site
-  Anticipated Elevated Radon
-  Sampling Points*
-  New Vegetation Sampling Area

*Sampling locations and spatial orientation based on MILDOS modeling and baseline gamma survey

New Vegetation and Surface Soil Sampling Locations

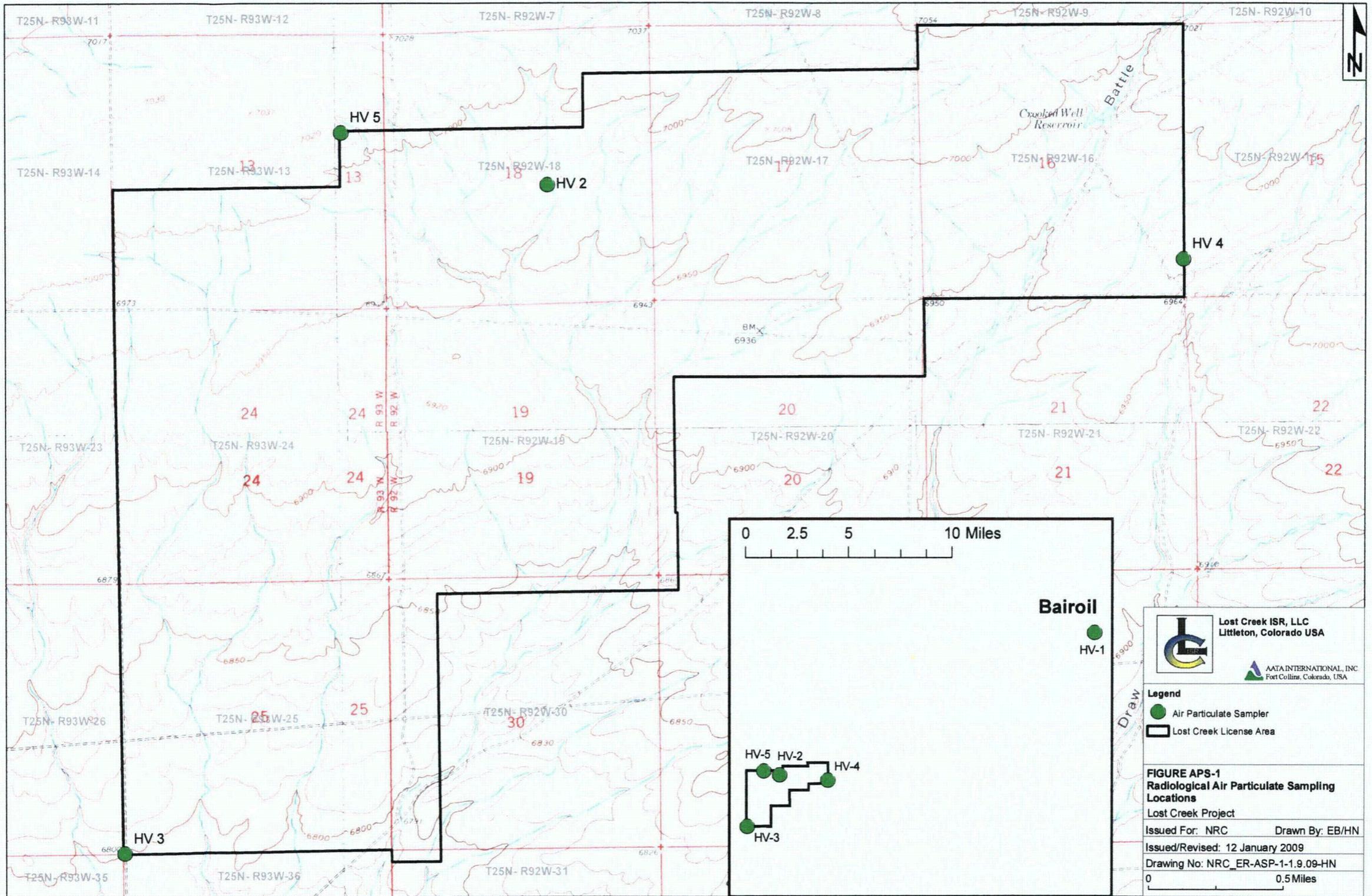
Lost Creek Project

Issued/Revised: 7.30.09 Drawn By: CF

0 190 380 760 Meters



ATTACHMENT 2



Lost Creek ISR, LLC
 Littleton, Colorado USA

AATA INTERNATIONAL, INC
 Fort Collins, Colorado, USA

Legend

- Air Particulate Sampler
- Lost Creek License Area

FIGURE APS-1
Radiological Air Particulate Sampling Locations
 Lost Creek Project

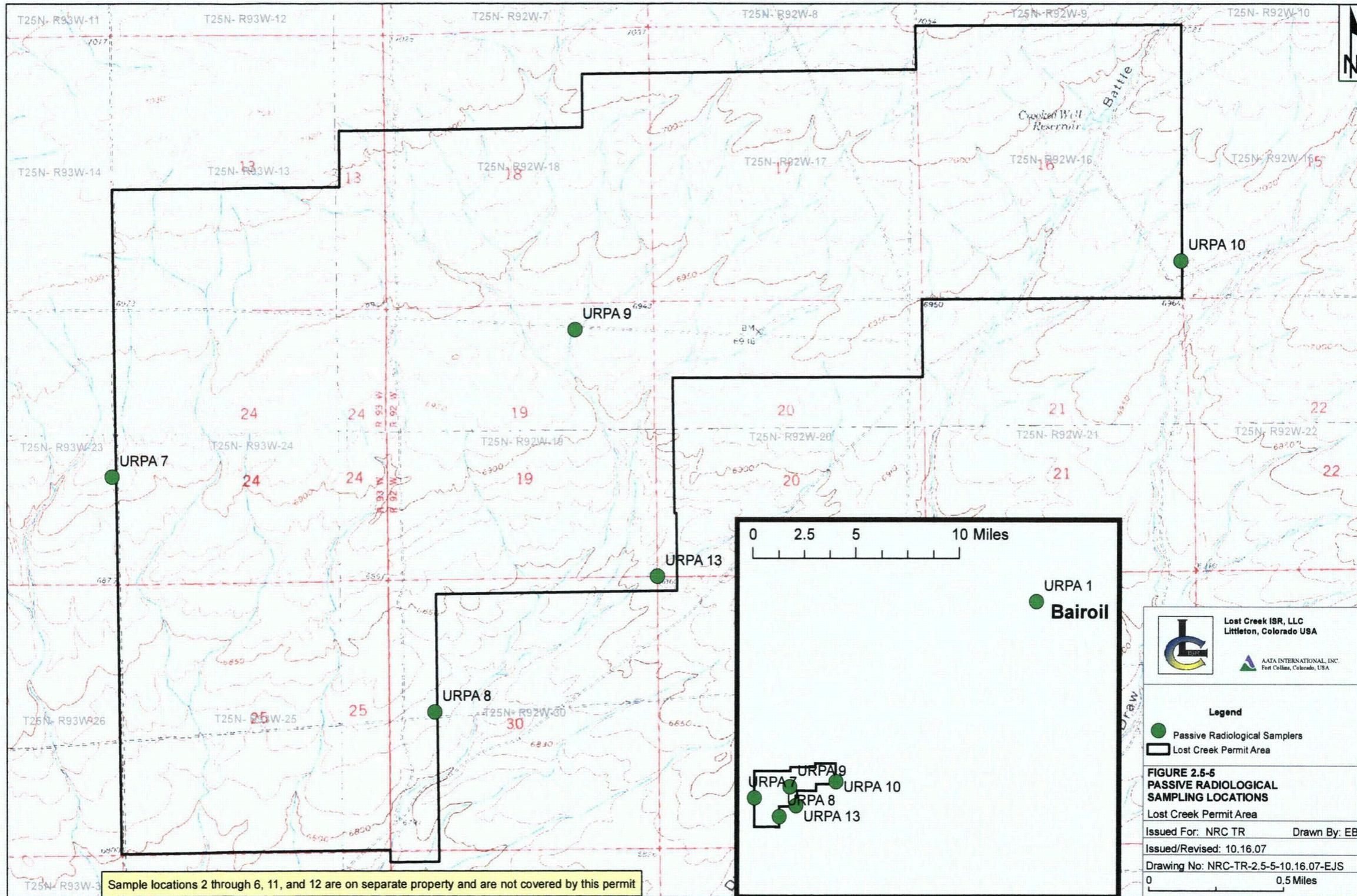
Issued For: NRC Drawn By: EB/HN

Issued/Revised: 12 January 2009

Drawing No: NRC_ER-ASP-1-1.9.09-HN

0 0.5 Miles

ATTACHMENT 3



Sample locations 2 through 6, 11, and 12 are on separate property and are not covered by this permit

Lost Creek ISR, LLC
Littleton, Colorado USA

AATA INTERNATIONAL, INC.
Fort Collins, Colorado, USA

Legend

- Passive Radiological Samplers
- ▭ Lost Creek Permit Area

FIGURE 2.5-5
PASSIVE RADIOLOGICAL SAMPLING LOCATIONS

Lost Creek Permit Area

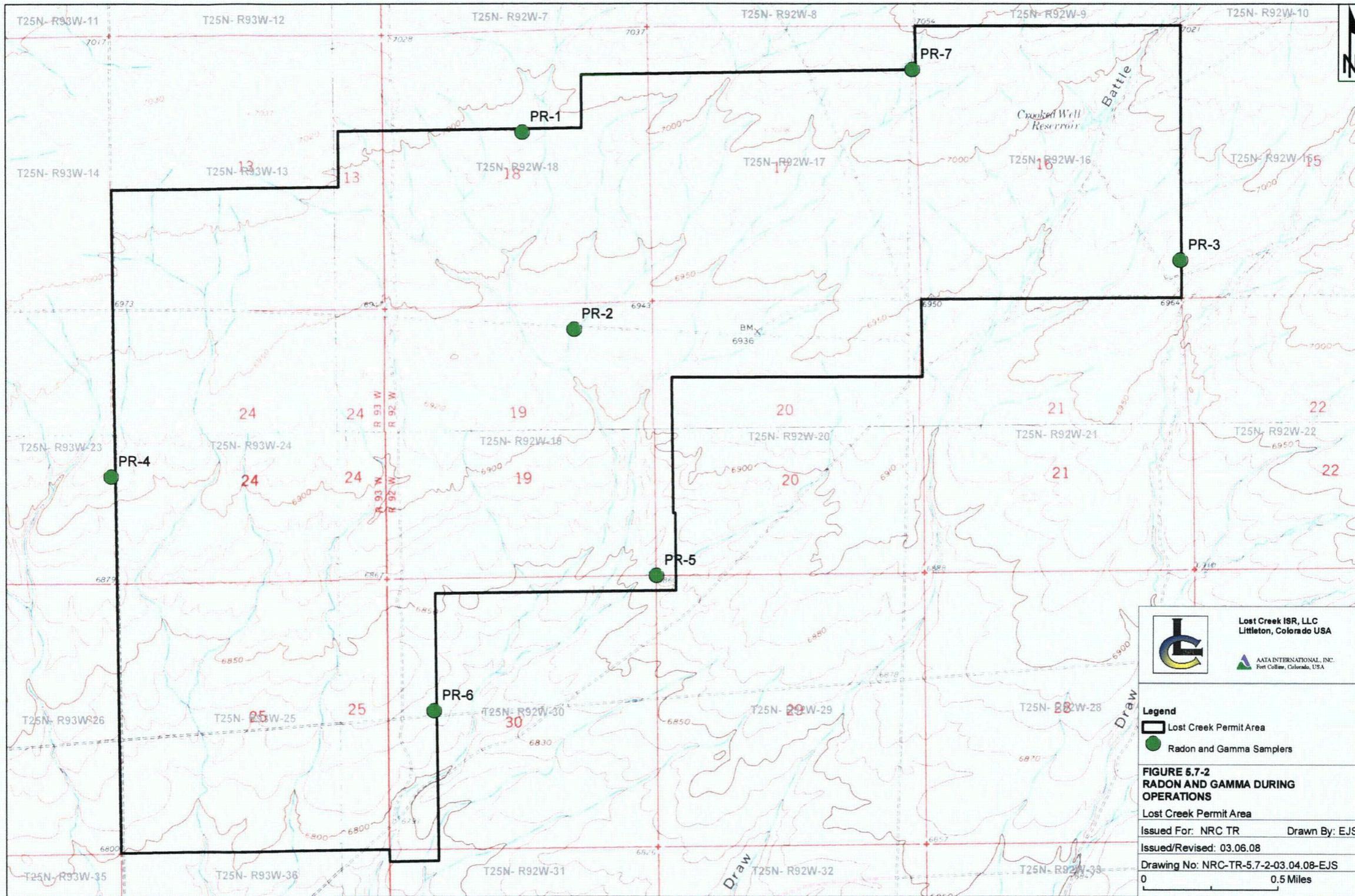
Issued For: NRC TR Drawn By: EB

Issued/Revised: 10.16.07

Drawing No: NRC-TR-2.5-5-10.16.07-EJS

0 0.5 Miles

ATTACHMENT 4




Lost Creek ISR, LLC
 Littleton, Colorado USA

AATA INTERNATIONAL, INC.
 Fort Collins, Colorado, USA

Legend
 Lost Creek Permit Area
 Radon and Gamma Samplers

**FIGURE 6.7-2
 RADON AND GAMMA DURING
 OPERATIONS**
 Lost Creek Permit Area
 Issued For: NRC TR Drawn By: EJS
 Issued/Revised: 03.06.08
 Drawing No: NRC-TR-5.7-2-03.04.08-EJS
 0 0.5 Miles