WYOMING MINING ASSOCIATION

July 23, 2009

Mr. Bill Von Till, Chief
Uranium Recovery Licensing Branch
Decommissioning and Uranium Recovery Licensing Directorate
Division of Waste Management and Environmental Protection
Office of Federal and State Materials and Environmental Management Programs
U.S. Nuclear Regulatory Commission
Washington, DC 20555–0001

Dear Mr. Von Till:

Subject: **Wyoming Mining Association - Comments on the April 23, 2009 Letter Entitled *SUMMARY OF APRIL 16, 2009 MEETING WITH LOST CREEK ISR, LLC***

The Wyoming Mining Association (WMA) is an industry association representing mining companies, contractors, vendors, suppliers and consultants in the State of Wyoming. Among its mining industry members are uranium recovery licensees, including in-situ and conventional uranium recovery operators, several companies planning new uranium recovery operations and several companies conducting final reclamation/restoration operations. WMA has reviewed the letter dated April 23, 2009 sent by NRC to Mr. Wayne W. Heili President of Lost Creek ISR, LLC. Because the letter's content has implications to the entire uranium recovery industry in Wyoming, WMA is providing the following comments on its content:

Discussion of Item 1: U3O8 / Accident scenarios

The letter states:

> *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.*

The term U3O8 is often used as "*percent U3O8*" for purposes of establishing a price for sale. For example, The UX Consulting Company, LLC. (http://www.uxc.com/) uses the term U3O8 as the material for establishing price. The products of interest are uranyl carbonate and uranyl peroxide, since initially the Lost Creek ISR product will not be dried.

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When eventually dried by a modern vacuum drier at relatively low temperature, no "U3O8" will be produced. In fact, the combination of 3 uranium atoms to 8 oxygen atoms is not possible given the valance states that are available for these elements. The term U3O8 is simply used to express the ratio of uranium and oxygen present in yellowcake. Hydrates of UO3 and Uranyl Peroxide are expected. Products shipped from uranium recovery facilities vary in color from yellow to orange yellow to dark green depending on the water of hydration and oxygen content of the material. These variations are caused by differing methods of drying (rotary vacuum versus calcining) and different methods of precipitation, (hydrogen peroxide, ammonia etc.). Undoubtedly these color variations represent differences in chemical composition. In general, a dark color is indicative of a lower valence state (+ VI, (UO3) through +IV (UO2)) and in this chemical phase the product is more insoluble.

NRC INFORMATION NOTICE 99-03:
EXOTHERMIC REACTIONS INVOLVING DRIED URANIUM OXIDE POWDER (YELLOWCAKE) discusses dried yellowcake precipitated with hydrogen peroxide at in-situ uranium recovery facilities stating:

> *On two separate occasions, licensees experienced expansion of the sealed drums from generation of oxygen from the decay of hydrogen peroxide. In the production process, hydrogen peroxide is used to precipitate uranium in the form of uranium oxide (U_XO_X). This precipitate is then pressed through a filter, dried in an oven and packaged in drums for shipment. Even though the dryer temperatures reached 537.7° C (1000°F), converting the precipitate to uranate peroxide ($UO_4 2H_2O$), sufficient hydrogen peroxide either remained in the product or was generated to develop oxygen pressure by the following reactions:*

$$UO_4 + 4H_2O \rightarrow UO_3 + H_2O_2 \uparrow + 3H_2O \uparrow$$
$$2H_2O_2 \rightarrow 2H_2O + O_2 \uparrow$$

In this case clearly the Commission states that the uranium compounds involved are UO4 and UO3. The two (2) incidents discussed in the quoted text above are isolated in nature. Large quantities of yellowcake product have been drummed without incident over a period of many years.

Determination of the actual chemical compounds of uranium being shipped from a given facility can only be determined via x-ray crystallographic analysis (e.g., X-ray diffraction techniques) of the material following commencement of operations.

Rather than debate the exact nature of the produced product, WMA believes that uranyl peroxide should be considered the product for an in-situ uranium recovery facility until commencement of operations at which point the initially produced material should be tested via x-ray crystallography in order to determine the actual composition of the product. Following that testing the actual composition based on the test results will be used. Should the process be changed the new post-change product should be tested and changes made based on the new test data if warranted. This is especially true in light of the fact that Regulatory Guide 8.30 *HEALTH PHYSICS SURVEYS IN URANIUM RECOVERY FACILITIES "in fact calls out UO_4 specifically stating:*

"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"

Additional information from the literature demonstrating the relative solubility of UO4 and UO3 is provided in subsequent sections of this letter. It should be noted that since UO4 is considered more soluble than the U3O8 used in the NUREG/CR – 6733 accident scenario, a similar accident analysis performed for modern ISRs would result in less worker/public dose since the UO4 product will have much less pulmonary retention as a TGLD Class D or W compound, relative to the much more insoluble U3O8 Class Y.

Discussion of Item 2: Derived airborne concentration (DAC)

The NRC letter to LCI **states:**

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.

Appendix 1 contains a discussion on uranium solubility and the applicable DAC. This discussion is excerpted from material provided by Mr. Steve Brown of SENES Consultants Limited. Mr. Brown is a Certified Health Physicist with over thirty years of experience in uranium health physics at NRC licensed and DOE nuclear weapons facilities. WMA requested the assistance of Mr. Brown in providing NRC with additional information related to the topics in the LCI letter.

WMA also understands that the Staff does not consider the following paper as being an acceptable source of information on yellowcake solubility:

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

This is disturbing in that this is a peer reviewed scientific paper published in the Journal of Health Physics. 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 using processes similar historical methods reported in the literature for determining solubility of uranium mill

products and to those used at existing and planned ISR facilities. WMA requests an explanation of the basis for the NRC Staff questioning the credibility of this study. (Note: this study was also included in the references provided by Mr. Brown – see Appendix I). If this peer reviewed study is being called into question, then reasonable arguments should be provided to support the assertions.

Discussion of Item 3: Worker dose calculations

The NRC letter to LCI states:

> *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.*

NUREG-1569 Standard Review Plan for In Situ Leach Uranium Extraction License Applications makes no mention of analysis for any isotope of thorium. It discusses uranium only stating:

> *Estimation of airborne uranium concentrations takes into account the maximum production capacity requested in the application and the anticipated efficiencies of airborne particulate control systems reviewed using in Sections 4.1 and 5.7.1 of this standard review plan.* (page 5-26) and;
>
> *Exposure calculations for natural uranium are consistent with Regulatory Guide 8.30, Section 3 (NRC, 2002).* (page 5-25)

Isotopes of thorium are not a major component of the dose at uranium in-situ recovery facilities. Work by Steve Brown, indicates that very little thorium is mobilized from the host formation by in-situ uranium recovery operations. Thorium is not mobilized to any extent by the lixiviant, is not selected by the IX resin and therefore does not need to be a consideration in establishing DACs or release limits. Please reference:

(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, September; 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)

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 particular, 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). NRC could have requested data to address this question from existing licensees.

Regulatory Guide 8.30 also discusses surveys for airborne yellowcake which are the only surveys required for an in-situ uranium recovery facility since ore dust is not present. It does not address other radionuclides, stating:

> *If the intake of such yellowcake is controlled to protect the kidney from the chemical toxicity of uranium, radiological protection criteria for natural uranium will also be satisfied. For purposes of compliance with 10 CFR Part 20, yellowcake undried or dried at low temperature should be classified as soluble.*

No licensee should be required to use anything other than the (DAC) for natural uranium unless concentrations of other radionuclides in quantities sufficient to impact worker dose are discovered in the product, in which case the provisions of 10 CFR §20.1204(g) would apply (e.g., use of the "sum of fractions rule" to establish a nuclide mix specific DAC). In this question, NRC Staff appears to be disregarding the experience gained by the ISR industry over the past 30 years as reflected in existing NRC guidance (e.g., NUREG 1569, Regulatory Guide 8.30). In doing so, they are implying that ISR operations are applying an inappropriate DAC since historical and existing NRC guidance is inadequate resulting in underestimation of worker exposures.

Discussion of Item 4: Contamination control program

The NRC letter to LCI discuses release limits for unrestricted use and states the following:

> *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.*

This is a very serious comment and it impacts both existing and future licensees. Limits for release for unrestricted use are clearly discussed and described in Regulatory Guide 8.30 and are not "ambiguous". These limits are discussed in further detail below

Table 5.7.6.3-1. Acceptable Surface Contamination Levels (U.S. Atomic Energy Commission, 1974)

Nuclides[a]	Average[b,c,d]	Maximum[b,d,e]	Removable[b,d,f]
Natural Uranium, Uranium-235, -238, and associated decay products	5,000 α dpm/100 cm^2	15,000 α dpm/100 cm^2	1,000 α dpm/100 cm^2
Transuranics, Radium-226, Radium-228, Thorium-230, Thorium-118, Protactinium-231, Actinium-227, Iodine-125, Iodine-129	100 dpm/100 cm^2	300 dpm/100 cm^2	20 dpm/100 cm^2
Natural Thorium, Thorium-232, Strontium-90, Radium-223, -224, Uranium-232, Iodine-126, Iodine-131, Iodine-133	1,000 dpm/100 cm^2	3,000 dpm/100 cm^2	200 dpm/100 cm^2
Beta-gamma emitters (nuclides with decay modes other than alpha emission or spontaneous fission) except Strontium-90, and others noted above	5,000 dpm/100 cm^2	15,000 dpm/100 cm^2	1,000 dpm/100 cm^2

[a] Where surface contamination by both alpha- and beta-gamma-emitting nuclides exists, the limits established for alpha- and beta-gamma-emitting nuclides should apply independently.
[b] As used in this table, dpm (disintegrations per minute) means the rate of emission by radioactive material as determined by correcting the counts per minute observed by an appropriate factor for background, efficiency, and geometric factors associated with the instrumentation.
[c] Measurements of average contamination should not be averaged over more than 1 m^2. For objects of less surface area, the average should be derived for each such object.
[d] The average and maximum radiation levels associated with surface contamination resulting from beta-gamma emitters should not exceed 0.2 mrad/hr at 1 cm and 1.0 mrad/hr at 1 cm, respectively, measured through not more than 7 mg/cm^2 of total absorber.
[e] The maximum contamination level applies to an area of not more than 100 cm^2.
[f] The amount of removable radioactive material per 100 cm^2 of surface area should be determined by wiping that area with dry filter or soft absorbent paper, applying moderate pressure, and assessing the amount of radioactive material on the wipe with an appropriate instrument of known efficiency. When removable contamination on objects of less surface area is determined, the pertinent levels should be reduced proportionally and the entire surface should be wiped. Reference: U.S. Atomic Energy Commission. Regulatory Guide 1.86, "Termination of Operating Licenses for Nuclear Reactors." Washington, DC: U.S. Atomic Energy Commission. June 1974.

NUREG-1569 Standard Review Plan for In Situ Leach Uranium Extraction License Applications states:

> *(9) 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 5.7.6.3-1: (page 5-31)*

The referenced table (originally from *Regulatory Guide 1.86 TERMINATION OF OPERATING LICENSES FOR NUCLEAR REACTORS – June 1974*) is included below:

Regulatory Guide 8.30 HEALTH PHYSICS SURVEYS IN URANIUM RECOVERY FACILITIES is far more specific when discussing release limits stating (Table 2 8.30-12):

TABLE 2

Table 2

Surface Contamination Levels for Uranium and Daughters on Equipment To Be Released for Unrestricted Use, on Clothing, and on Nonoperating Areas of UR Facilities*

Average**	5,000 dpm alpha per 100 cm2 Average over no more than 1m2
Maximum**	15,000 dpm alpha Applies to an area of not more than 100 cm2 per 100 cm2
Removable	1,000 dpm alpha per 100 cm2 Determined by smearing with dry filter or soft absorbent paper, applying moderate pressure, and assessing the amount of radioactive material on the smear

* These values are taken from Regulatory Guide 1.86, "Termination of Operating Licenses for Nuclear Reactors" (Ref. 23), 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," Division of Fuel Cycle and Material Safety, USNRC, Washington, DC 20555, August 1987 (Ref. 24). Available in NRC Public Document Room for inspection and copying for a fee.
** The value includes both fixed and removable contamination.
(The contamination levels in Table 2 are given in units of dpm/100 cm2 because this is the minimum area typically surveyed. When performing a smear or wipe test, the area should roughly approximate 100 cm2. However, there is no need to be precise about the area to be smeared.)

This release guidance is abundantly clear. The letter states:

> *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.*

It is not industry that is *including radium (and potentially thorium-230) in the gross alpha measurements* but the NRC in its own guidance. Both radium-226 and thorium-230 are daughters (decay products) of natural uranium and are included in the above described release limits. It is not appropriate to attempt to change release limits for an entire industry that are part of published/final regulatory guidance in a letter to a single applicant. It appears that Commission staff is requesting that a prospective licensee propose methods contrary to in place/approved standards of practice. Regulatory guidance exists to serve both the applicant and the application reviewer. If the applicant cannot rely on existing guidance in order to prepare a license application and the reviewer considers the guidance ambiguous and does not use it, the license application process breaks down.

The release limits in *Regulatory Guide 8.30 HEALTH PHYSICS SURVEYS IN URANIUM RECOVERY FACILITIES* Table 2 are for "*uranium and daughters*". The use of "*uranium and daughters*" is entirely appropriate for uranium recovery facilities as discussed below by Mr. Brown:

> *I believe it is also important to note that at U mills and ISRs, Ra 226 in total absence of other uranium series radionuclides is almost impossible – including in ISR precipitates and filtrates from ISRs. The inclusion of "Ra 226, Ra 228" in the more restricted contamination category with the transuranics was intended for facilities/activities that used pure radium and/or its salts (e.g. radium needles in medical applications, luminous dial manufacturers) which resulted in much higher biological availability to potentially exposed workers in those industries but is inappropriate and unnecessary when much less biologically available forms and/or other much lower toxicity uranium series nuclides are present.*

In my personal experience as an RSO at numerous uranium fuel cycle facilities, including ISRs, I was subject to numerous health physics inspections by NRC staff. The general understanding always was that at any uranium facility involving natural enrichment and associated uranium series decay products, in any combination, the contamination limits of Regulatory Guide 1.86 for "U-nat, U-235, U-238, and associated decay products" were applicable and adequately protective. When I first started in the U industry in 1976, pre dating RG 1.86, the same limits appeared in a NMSS document entitled "Annex A" and then similar title to RG 1.86".

In my personal experience at USDOE nuclear facilities and at USACE FUSRAP environmental restoration projects, identical contamination limits of Regulatory Guide 1.86 for "U-nat, U-235, U-238, and associated decay products" were similarly considered applicable and adequately protective.

Applying contamination limits applicable and appropriate for transuranics, i.e. nuclear weapons facilities and laboratories, ("Transuranics, Ra-226, Ra-228, Th-230, Th-228, Pa-231, Ac-227, 1-125,1-129 ") to uranium recovery facilities does not appear to be appropriate or necessary in the interest of worker / public protection, in consideration of practicality of measurement and assessment (interference from other uranium series nuclides other than just "Ra 226") and taking into account "social and economic factors" as required by ALARA. An appropriate and proper "standard of care" is provided by the historical interpretation and application of the RG 1.86 limits in the category "U-nat, U-235, U-238, and associated decay products" at uranium recovery facilities including ISRs.

Uranium recovery licensees have for years used the release limits described in Table 2 above. The following documents have been provided to existing licenses or attached to their licenses:

Policy and Guidance Directive FC 83-23: Termination of Byproduct, Source and Special Nuclear Material Licenses – November 4, 1983

Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct or Source Materials – September 1984

Attempting to alter release limits at this point would call in to question the release for unrestricted use of structures and/or items from licensed and/or formerly licensed uranium recovery facilities in the past and previous clearance / releases to the public domain of formerly owned Federal sites (or portions thereof) with TENORM histories. Many decisions regarding environmental restoration clearance, radioactive waste management and license termination across the United States used the historical interpretation of "*Natural Uranium, Uranium-235, -238, and associated decay products*". Large amounts of material including heavy equipment/machinery owned by third party contractors have been released under the limits included in Table 2 above. Altering these limits with no basis may cause owners of previously released items to question the validity of the release, potentially creating serious problems between licensees and their contractors.

The release limits of Regulatory Guide 1.86, including the historical and universally accepted applicability of the category "*"U-nat, U-235, U-238, and associated decay products*" are protective , provide for an adequate "standard of care" and have been accepted as applicable at NRC, DOE, USACE and other TENORM sites across the US for 30 years and continued use at uranium in-situ recovery operations is consistent with existing and approved NRC guidance.

Dr. Robert Meyer of Tetra Tech who is the Radiation Safety Officer (RSO) of record at three (3) remedial action sites provided the following discussion regarding release for unrestricted use:

> *I am currently the RSO of Record at three remedial action sites (in Wyoming and Texas). Free releases of gear including heavy equipment have occurred at all of the sites, in all cases using the limits noted in the Regulatory Guide 1.86 table referenced in the subject letter. Formal license inspections have occurred at all three sites. During those inspections, the responsible agency representatives (USNRC or Texas) have verified compliance with applicable standards. These inspectors have specifically examined the free release procedures and standards we employed. In all cases, no findings of any type resulted from these official inspections, nor were any concerns expressed with regard to our free release methods or standards. Our policy for free release from these sites is ALARA, and in fact equipment is generally released at essentially background fixed and removable surface activities, but the actual limits specified for allowable release are as noted in the Regulatory Guide 1.86 free-release table.*

In addition, during the course of the Uranium Mill Tailings Remedial Action Project based in Albuquerque, New Mexico, 22 Title 1 uranium mills or tailings piles were remediated. The remedial work included free release, per the standards exemplified in Regulatory Guide 1.86, of many thousands of items of equipment including heavy trucks, graders, dozers, trailers, backhoes, railcars (used to move the Utah Vitro site tailings) and other large items. The UMTRA Project was probably the largest uranium-tailings-related remedial action project in U.S. history. The project was inspected by representatives of cognizant State and Federal (USNRC, USDOE and USEPA) agencies. Use of the free release Standard Operating Procedures (SOPs), based on the free release standards noted in the subject letter, was never questioned by any agency representative or other auditing authority. In discussing release criteria, Mr. Brown states:

> *It should be noted that the 1.86 limits, including the general understanding that "U-nat, U-235, U-238, and associated decay products" means natural uranium series radionuclides, in various combinations and equilibrium states has been used throughout the nuclear industry for over 30 years (not just by NRC licensees, also by DOE and the USACE.). Examples include:*
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> *Regulatory Guide 8.30 Health Physics Surveys in Uranium Recovery Facilities – Table 2 reproduces exactly the RG 1.86 table 1 values. Additionally, Section 4.7, Contamination of Skin and Clothing specifies "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"*
>
> *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 Intuitions – Table 3 same as RG 1.86, Table 1*
>
> *US DOE Order 5400.5 – Radiation Protection of the Public and Environment – Figure IV – 1 identical to Table 1, Regulatory Guide 1.86*

*NUREG–5849, "Manual for Conducting Radiological Surveys in Support of License Termination" (Berger, 1992) and NUREG–1575, Revision 1, "Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM) also both use the surface contamination limits identical to Regulatory Guide 1.86 table 1. Many of the Federal sites remediated under these guidelines were contaminated with TENORM (natural uranium series nuclides) and to the best of my knowledge and belief the "**U-nat, U-235, U-238, and associated decay products**" limits were consistently applied across the US in those cases.*

NRC staff references Fuel Cycle Memorandum 83-23, "Termination of Byproduct, Source and Special Nuclear Material Licenses" (1983?). This historical document provided guidelines for acceptable average and maximum surface contamination levels for a wide variety of radionuclides. It also provided average and maximum radiation levels of 0.2 and 1.0 millirad per hour at 1 centimeter for beta and gamma-emitters. Its Table 1 also has the category for "U-nat, U-235, U-238, and associated decay products" as specified in Regulatory Guide 1.86 and uses the identical average contamination limit of 5000 dpm / 100cm2.

A previous attempt was made to revise release limits. On June 30, 1999 in Federal Register / Vol. 64, No. 125 / Wednesday, June 30, 1999 / Proposed Rules page 35090 the NRC released *Release of Solid Materials at Licensed Facilities: Issues Paper, Scoping Process for Environmental Issues, and Notice of Public Meetings.* In this document the NRC discussed release criteria and Regulatory Guide 1.86 and postulated a series of questions for discussion in written comments and in public meeting.

This effort is discussed in *NUREG-1761 - Radiological Surveys for Controlling Release of Solid Materials Draft Report for Comment* which states:

On June 30, 1999, the NRC published, for public comment, an issues paper indicating that the agency was examining its approach for control of solid material. The issues paper presented alternative courses of action for controlling the release of solid materials that have very low amounts of, or no, radioactivity.

In August 2000, the Commission decided to defer its final decision on whether to proceed with rulemaking on controlling the release of solid materials while it requested a study by the National Academies on possible alternatives for controlling the release of slightly contaminated materials.

On June 1, 2005, the Commission unanimously disapproved proceeding with the recommendations of *SECY-05-0054: PROPOSED RULE: RADIOLOGICAL CRITERIA FOR CONTROLLING THE DISPOSITION OF SOLID MATERIALS.*

The WMA believes that it is inappropriate for NRC staff to attempt to revise release standards via a letter to an applicant for a source material license while revision of existing regulations on release have been deferred pending further study by the Commission. It seems that if NRC is determined to revise guidance that they should start the process by collecting data at operating facilities to ensure that any proposed changes are in fact risk informed.

The NRC letter to LCI also states:

> *NRC staff examples include dose calculations in SECY 98-155 and release criteria for the Molycorp York, PA facility.*

Molycorp's York, Pennsylvania facility was a rare earth processing facility. It is discussed in the United States Geological Survey (USGS) document *RARE EARTHS* By James B. Hedrick as follows:

> *Molycorp continued to decommission and decontaminate its closed rare-earth processing facilities at Washington and York, PA. Limited amounts of naturally occurring low-level radioactive material (thorium) were planned for removal to approved disposal sites.*

The site processed materials extracted from bastnasite ores from Molycorp's Mountain Pass mine, near Mountain Pass, California. Bastnasite is described as follows:

> *There is bastnäsite-(Ce) with a formula of (Ce, La)CO_3F. There is bastnäsite-(La) with a formula of (La, Ce)CO_3F. There is also bastnäsite-(Y) with a formula of (Y, Ce)CO_3F. Most bastnäsite is bastnäsite-(Ce), and cerium is by far the most common of the rare earths in this class of minerals.* (Wikipedia)

The bastnasite from the Mountain Pass Mine was predominately thorium bearing basnasite as described below:

> For example, thorium-bearing bastnasite occurs mechanically mixed with barite as major constituents of the ore from the Mountain Pass Mine in San Bernardino County, California. (*Process for obtaining permeability logs using radioactive drilling mud additives United States Patent 4691772*)

Application of the 100/300 dpm/100 cm2 total and 20 dpm/100cm2 removable limit for thorium-230 or the 1000/3000 dpm/100 cm2 total and 200 dpm/100 cm2 removable limits for natural thorium / thorium-232 may have been entirely appropriate for Molycorp's facility given that the primary radioactive element of concern was natural thorium. However, application of these limits is inappropriate for uranium recovery facilities.

The NRC letter to LCI also states the following:

> *The result may be that LCI cannot release any equipment until specific isotopes and their associated release limits are evaluated.*

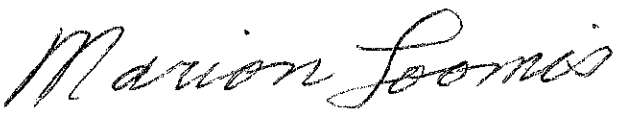
WMA has a number of concerns with this statement. First, this stipulation would clearly apply to existing uranium recovery licensees that have been and are currently using the release limits specified in Regulatory Guide 8.30 Table 2 for natural uranium and its daughters to release materials and equipment. Second, who will evaluate the release limits for specific isotopes and how long will this evaluation take? The discussion above clearly documents the historical, unambiguous and in general, universal application of the Regulatory Guide 1.86 category of *"U-nat, U-235, U-238, and associated decay products"* at TENORM sites and licensed uranium recovery facilities. Accordingly, this "concern" should be satisfied.

WMA believes that any changes to release limits from those in Regulatory Guide 8.30 only be done as part of the release of solid materials regulatory process which is open to public comment and not via a letter to a single applicant.

In summary, WMA is concerned that NRC staff appears to be unilaterally dismissing radiological protection approaches for uranium recovery facilities that are based on 30 years of experience gained by NRC and the industry. It is unclear to industry if the NRC's justifications for dismissing long standing, vetted practices and guidance are risk informed. In connection with the letter to LCI, Staff has stated that the guidance provided in NUREG-1569 and Regulatory Guide 8.30 is incorrect and has questioned the scientific validity of peer- reviewed studies. The implication is that the industry, in following this guidance, is underestimating worker exposures and releasing materials in excess of appropriate limits. WMA members take their responsibilities for radiological protection very seriously and have worked with NRC over the years to develop the existing guidance. WMA members stand ready to continue that collaboration with NRC to ensure the safety of workers and the public. To that end, this letter has provided an extensive list of existing literature relevant these issues. WMA believes that questioning mature standards of practice and existing radiological protection programs and guidance in a letter to a single applicant is not an appropriate approach.

The Wyoming Mining Association (WMA) appreciates your consideration of these comments on the letter received by one of its uranium recovery license applicant members. If you have any questions please do not hesitate to contact me.

Sincerely yours,
WYOMING MINING ASSOCIATION



Marion Loomis
Executive Director

Cc: Katie Sweeney – National Mining Association (NMA)

Appendix 1

The following is a discussion with references provided by Mr. Brown for use in this letter:

*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). It appears that LCI is arguing that the UO_4 product should be Task Group on Lung Dynamics (TGLD - ICRP 19) class D or W (most or moderately soluble – equivalent to ICRP F – or M - fast or medium dissolution), NRC staff suggests justification is not adequate 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 establishing air monitoring parameters, for worker airborne exposure control and dose assessment.*

A few specifics in support of the Class D or W designation for $U0_4$:

- *RG 8.30 in fact calls out U04 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 my general list below.

- *See rReference: 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 UO4 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 in my 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 regarding the question of UO4 solubility, Ron responded as follows: "*UO4 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 UO4". (NOTE: See the Kathren and Burklin reference above).

- *Reference: Solubility Characteristics of Airborne Uranium From and In Situ Uranium Processing Plan. Metzger R, Wichers D. et al. Health Physics 72.3, March 1997 p 418 – based on in vitro solubility studies of a UO_4 product, dissolution rates are demonstrated clearly indicative of class D (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 compound. See Task Group on Lung Dynamics* (TGLD) classification system of D, W and Y in ICRP 19, " Metabolism of the Compounds of Plutonium and Other Actinides", 1972.

Examples of some of the numerous studies and references published over the last 30 + years that specifically address solubility and solubility class of uranium mill and other 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. *Agency for Toxic Substances and Disease Registry (ATSDR). 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. *Alexander RE. Biokinetics model for uranium inhalation/excretion, uranium mill workers. 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, 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. *Alexander, R. **E.**, "Applications of Bioassay for Uranium," WASH-1251, * U.S. Atomic Energy Commission, Washington, DC, 1974.*
11. *Spitz, H. B.. **J.** C. Simpson, and T. L. Aldridge (Pacific Northwest Laboratory, Battelle Memorial Institute), "Analysis of Uranium Urinalysis and In Vivo Measurement Results from Eleven Participating Uranium Mills," NUREG/CR-2955, * U.S. Nuclear Regulatory Commission, Washington, DC, 1984.*
12. *D. R. Kalkwarf, "Solubility Classification of Airborne Products from Uranium Ores and Tailings Piles," NUREG/CR-0530, USNRC, January 1979.*
13. *F. Eidson and J.A. Mewhinney, "In Vitro Dissolution of Uranium Product Samples from Four Uranium Mills," NUREG/CR-0414, USNRC, October 1978.*

Appendix 2

References

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USNRC, "Guidelines for Decontamination of facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct or Source Materials," September 1984

USNRC, "INFORMATION NOTICE 99-03: EXOTHERMIC REACTIONS INVOLVING DRIED URANIUM OXIDE POWDER (YELLOWCAKE)," January 29, 1999

USNRC, "NUREG-1569 - Standard Review Plan for In Situ Leach Uranium Extraction License Applications Final Report - June 2003

USNRC, "NUREG-1761- Radiological Surveys for Controlling Release of Solid Materials Draft Report for Comment," July 2002

USNRC, "Policy and Guidance Directive FC 83-23: Termination of Byproduct, Source and Special Nuclear Material Licenses," November 4, 1983

USNRC, "REGULATORY GUIDE 8.30 - HEALTH PHYSICS SURVEYS IN URANIUM RECOVERY FACILITIES," May 2002

USNRC, "REGULATORY GUIDE 1.86 - TERMINATION OF OPERATING LICENSES FOR NUCLEAR REACTORS," June 1974

USNRC, "Release of Solid Materials at Licensed Facilities: Issues Paper, Scoping Process for Environmental Issues, and Notice of Public Meetings," June 30, 1999

USNRC, "SECY-05-0054: PROPOSED RULE: RADIOLOGICAL CRITERIA FOR CONTROLLING THE DISPOSITION OF SOLID MATERIALS" June 1, 2005

USNRC, "SUMMARY OF APRIL 16, 2009 MEETING WITH LOST CREEK ISR, LLC," April 23, 2009

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