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5.0 OPERATIONAL ORGANIZATION, MANAGEMENT, PROGRAMS, & TRAINING

5.1 Corporate Organization and Administration

Management and operation of monitoring programs at the Project are the responsibility of the six positions within LC ISR, LLC. Those positions are:

1) President;
2) General Manager;
3) Mine Manager;
4) Manager of Environment, Health, and Safety (EHS) and Regulatory Affairs;
5) Site Supervisor EHS/Radiation Safety Officer (RSO); and
6) Department Heads: Site Chief Geologist, Project Engineer, and Operations Manager.

The organization of these positions is depicted in **Figure 5.1-1**. In addition, per the requirements of the NRC, a Safety and Environmental Review Panel (SERP) will be established to integrate the various roles that support the operation and maintenance of the mine.

All LC ISR, LLC employees responsible for day-to-day operation of the facility will spend the majority of their time at the mine site. These include the positions of Mine Manager, Site Supervisor Environmental Health and Safety/Radiation Safety Officer (EHS/RSO), Health Physics Technician, Department Heads, and Uranium Recovery Workers. The positions of President, General Manager, and Manager EHS and Regulatory Affairs will generally work out of the Casper, Wyoming office. LC ISR, LLC is a 100% owned subsidiary of Ur-Energy USA Inc. The President of LC ISR, LLC also serves in a senior management position with Ur-Energy USA Inc.

The Site Supervisor EHS/RSO will report directly to the Manager EHS and Regulatory Affairs and neither position shall have direct production responsibilities. This will allow the individuals filling these two positions to make sound decisions regarding EHS and radiation safety without being unduly pressured by production concerns. The Site Supervisor EHS/RSO shall have complete authority and responsibility to halt any work which they deem unsafe. The Mine Manager shall also have the authority and responsibility to halt any work which may be unsafe. No employee, regardless of position, shall place production ahead of protection of employees, the public, and the

environment. No employee will be forced to perform work which, in their view, is unsafe or could result in an unnecessary exposure to radiation.

Members of the Safety and Environmental Review Panel (SERP) shall be allowed to consider proposed actions without undue influence from senior management. In other words, the result of the SERP will not be a foregone conclusion, but will be based on a thorough review of the proposal and its merit.

The successful construction and operation of an in situ recovery facility relies on the close cooperation of several groups. Construction of the facility will be managed by the Engineering Group under the leadership of the Project Engineer. While the facility is being designed and constructed, members of the EHS Department will be regularly involved to ensure EHS and radiation issues are being properly addressed. During construction, daily safety inspections will be performed by the EHS Department to ensure construction activities are being performed in a manner consistent with regulations. The Engineering Group shall perform regular inspections during construction to ensure the facility is constructed in a manner consistent with the approved design. During operations, all significant projects and maintenance will be performed under a work order system. The work order system will include a mechanism for convening a SERP if warranted. All work orders will be reviewed by the Engineering Group and EHS Department before initiation of work to ensure the proposed work is technically sound and protective of employees, public and the environment.

5.1.1 President

The President maintains the ultimate responsibility for all operations of LC ISR, LLC, including the Project and its activities. This individual is responsible for interpreting and acting upon the Managers' policy and procedural decisions. The President has the responsibility and authority to immediately suspend, postpone, or modify an action deemed threatening to human health or the environment or deemed in violation of state and federal regulations.

To provide direction to Project management and employees, the President shall develop and publish a policy that defines the organization's commitment to protection of the environment, and health and safety of employees and the public. The President shall periodically review the policy to ensure continued relevance.

5.1.2 General Manager

The General Manager is responsible for the safe operations of LC ISR, LLC, including the Project. The General Manager reports directly to the President. The General Manager, with the assistance of the Manager of EHS and Regulatory Affairs, shall perform and document an annual review of the Environment, Health, and Safety Management System (EHSMS). The purpose of the annual review shall be: to ensure appropriate organization of the management system; to ensure adequate resources are available to protect the health and safety of employees, the public, and the environment; and to ensure trends in regulatory and/or policy noncompliance are recognized and root causes are mitigated.

5.1.3 Mine Manager

The Mine Manager for the Project reports to the General Manager, and is responsible for implementing managerial and financial actions that affect mining operations, and the EHSMS of LC ISR, LLC. This includes implementation of the company health and safety, radiation safety, environmental compliance, and licensing programs; and ensures that they are conducted in a safe and financially responsible manner, while maintaining compliance with applicable regulations, license conditions, and corporate policy. The Mine Manager will assist in the development, administration, and enforcement of the mining safety, radiation protection, environmental, and operational monitoring programs. These programs involve the development, review, approval, and implementation of mining process and safety protocols, as well as technical review, evaluation, and participation in routine audits of QA/QC programs. The Mine Manager also provides technical guidance and assistance concerning mining operations and activities to site personnel; including development and administration of corporate radiation protection programs and applicable mine safety programs. The Mine Manager has the responsibility and authority to immediately suspend, postpone, or modify an action deemed threatening to human health or the environment or deemed in violation of state and federal permitted and licensed regulations.

5.1.4 Manager of EHS and Regulatory Affairs

The Manager of EHS and Regulatory Affairs will have the responsibility and authority for LC ISR, LLC's radiation safety, environmental compliance, and quality assurance programs at the Project and LC ISR, LLC's other development activities. This person will provide input to the Site Supervisor EHS/RSO to ensure that all radiation safety, environmental compliance and permitting/licensing programs are conducted in a responsible manner, and in compliance with all applicable regulations and permit/license

conditions. The Manager of EHS and Regulatory Affairs reports directly to the General Manager. The Manager of EHS and Regulatory Affairs and shall assist the General Manager in the annual review and resulting documentation of the EHSMS as defined above.

5.1.5 Site Supervisor EHS/RSO

The Site Supervisor EHS/RSO reports to the Manager of EHS and Regulatory Affairs and is responsible for the daily supervision of the EHS Management System of the Project. This individual's responsibilities will include developing and implementing safety and environmental programs, properly maintaining and retaining records, and assisting the mine staff to comply with regulations and license conditions applicable to employee health protection. In addition, the Site Supervisor EHS/RSO will play an integral role in facility design and start-up and will work closely will all parties during the development of procedures, training, and inspections to ensure the As Low As Reasonably Achievable (ALARA) principle is adhered to throughout the Project.

The Site Supervisor EHS/RSO also is/will:

- the designated Site QA/QC Coordinator;
- a member of the As Low As Reasonably Achievable (ALARA) Committee and required to assist management with the annual ALARA Audit;
- required to report to the Manager of EHS and Regulatory Affairs and the Mine Manager on all matters regarding environmental protection and radiation and worker safety;
- conduct routine training programs for the supervisors and employees with regard to the proper application of radiation protection, emergency response, and environmental control programs;
- inspect the facilities to verify compliance with all applicable radiological health and safety requirements and the QA/QC program;
- annually review all operating procedures to ensure that radiation exposures will be maintained ALARA;
- authorized to terminate immediately any activity that may be a threat to the employees, public health, or the environment;
- coordinate implementation of the health physics programs with other departments within the facility to ensure compliance with regulations;
- responsible for ensuring that all health physics samples and records are complete, accurate, and properly filed and stored;
- responsible for routinely auditing all operational and monitoring procedures and the QA/QC programs;
- responsible for administering the Radiation Safety Program;

- monitor relative attainment of radiation exposure ALARA;
- receive 40 hours of applicable refresher radiation safety training from qualified instructors on a biennial basis;
- ensure that any non-routine work not covered by an SOP will be conducted in accordance with an RWP (Radiation Work Permit) as reviewed;
- ensure that the use, handling and transport of radioactive materials is restricted to qualified individuals that have received all proper training and approval from the RSO to perform these functions;
- make certain that transport of any radioactive materials generated at the site complies with all state and federal regulatory requirements for transport of applicable radioactive materials;
- ensure that all employees wear approved personal dosimetry radiation monitoring badges in areas as required;
- provide an annual dose report to all monitored individuals;
- coordinate and implement the calibration and maintenance of site radiation detection and survey instruments with the manufacturer at intervals recommended by the manufacturer, and ensure that all radiation survey instruments are in current calibration and proper working condition;
- ensure that all site personnel have read, understand and comply with all radiation safety program requirements;
- assist Department Heads with the development and revision of SOPs; and
- maintain the EHSMS, including SOPs, in such a manner that all employees have access to the most recent information regarding all relevant facets of environmental, health, and safety issues.

5.1.5.1 Health Physics Technician

The Health Physics Technician (HPT) shall assist the RSO with the implementation of the radiological safety program by collecting, documenting, and interpreting data. The HPT shall also help maintain radiation safety equipment such as survey meters. The HPT reports directly to the RSO.

5.1.6 Department Heads

The Department Heads include the Site Chief Geologist, the Project Engineer, and the Operations Manager. They are responsible for the site's operational and maintenance activities and procedures. Department Heads shall review the tasks that their respective employees will be performing and develop, with input from the Site Supervisor EHS/RSO, SOPs for any task that may present a hazard to the employee, public, environment, or operation. Department Heads will subsequently use the SOPs as training documents to ensure employees receive consistent and thorough training. Department

Heads shall enforce compliance with all facets of the EHSMS, including SOPs, in order to minimize risks. Department Heads or their designee shall perform and document an annual review of each SOP within their area to ensure continued accuracy and relevance. These individuals report to the Mine Manager. Development and review of procedures involving radiological safety concerns will be coordinated with the Site Supervisor EHS/RSO.

The construction of the facility and the subsequent integration of the construction activities with operations will be overseen by the Engineering Staff which is managed by the Project Engineer.

5.1.7 Uranium Recovery Workers

Because a radiation protection and ALARA program is only as effective as the workers' adherence to the program, all workers at the facility, especially those involved in daily uranium processing activities such as Plant and Mine Unit Operators and maintenance crews, will be responsible for:

- adhering to all rules, notices, and operating procedures for radiation safety established by management and the RSO;
- reporting promptly to the RSO any license management equipment malfunctions or violations of standard practices or procedures that could result in increased radiological hazard to any individual; and
- suggesting improvements for the radiation protection program to ensure it is ALARA.

5.2 Management Control Program

5.2.1 Environmental, Health, and Safety Management System (EHSMS)

In order to provide the highest level of protection to employees, the public, environment, and operation, site management (including the Manager of EHS and Regulatory Affairs, Mine Manager, Site Supervisor EHS/RSO, and Department Heads) shall develop and implement an Environmental, Health, and Safety Management System (EHSMS). Critical aspects of the management control program include: Standard Operating Procedures (SOPs) applicable in those situations routinely encountered; RWPs applicable in unanticipated conditions; recordkeeping to ensure all aspects of the EHSMS and the

effectiveness of the program can be evaluated by LC ISR, LLC and NRC; reporting to ensure that no part of the program is inadvertently overlooked; and Quality Assurance Program applicable to all key components of the various phases of an ISR project.

Standard Operating Procedures

An important aspect of the EHSMS is the development of SOPs. SOPs will be developed for all routine tasks which may present a hazard to employees, the public, environment, or the operation. Written health physics SOPs shall be developed for personnel and area monitoring, sampling, analysis, and instrument calibration. Department Heads shall be responsible for initiating the development of SOPs for all routine tasks within their area that may generate a hazard. The Site Supervisor EHS/RSO shall assist with the development of SOPs and may also initiate SOPs when the need arises. SOPs may be considered final when they have been approved by the respective Department Head and the Site Supervisor EHS/RSO. SOPs developed in response to SERP findings must be approved by the SERP before implementation. SOPs related to handling, processing, storing, or transporting radioactive materials and all health physics related SOPs will be annually reviewed by the RSO. The RSO and the management team will be responsible for seeing that employees are trained and provided guidance to ensure adherence to SOPs. Hard copies of SOPs will be readily accessible to personnel at work areas and will be part of the radiation safety training (**Section 5.5**) and on-the-job training.

Hard copies of the SOPs for specific work areas will be provided as controlled copies. The supervisor EHS/RSO will ensure that the controlled copies of the SOP Manuals are up-to-date and that no out-of-date SOPs are present in the active Manuals. The Supervisor EHS/RSO will be the custodian of the master copy of all SOPs and a complete controlled copy of the Procedures Manual.

Radiation Work Permits

Non-routine procedures or maintenance activities that may result in significant occupational exposure to radioactive materials and for which no SOP exists will be performed in accordance with an RWP. An RWP will be issued by the RSO or the HPT. Each RWP will, at a minimum, describe the:

- scope of work to be performed;
- estimated worker exposures;
- precautions necessary to reduce radiation exposure;
- necessary supplemental radiological monitoring and sampling prior to, during, and following completion of the work;
- RWP-specific training required; and
- personal protective equipment (PPE) required.

All RWPs will be reviewed and approved prior to performing the non-routine activities, and will be documented and filed. After completion of the Project for which the RWP was issued, the RSO will terminate the RWP. Throughout the duration of the license, records will be kept and retained concerning:

- materials control and tracking,
- the Radiation Safety Program,
- sampling,
- the survey and calibration programs,
- planned special exposures,
- occupational and public dosimetry tracking,
- disposal source, and
- decommissioning.

Recordkeeping

Records will be maintained as original hard copies, as copies on microfiche, or will be electronically protected, and will be readily retrievable for regulatory inspection. Records may be transferred to a new owner or licensee in the event that the property or license is transferred. Records may also be transferred to NRC after license termination.

LC ISR, LLC will track, control, and demonstrate control of the source and byproduct material at the site, such that on-site and off-site dose limits will not be exceeded. For the period identified in the license conditions or until license termination, records will be maintained and retained relating to the receipt, transfer, and disposal of any source or byproduct material processed or produced. In an identifiable, separate file, decommissioning records will include documentation of spills and unusual occurrences involving the spread of contaminants, the cleanup actions taken and the location of remaining contaminants.

With adequate safeguards against tampering and loss, the following records will be permanently maintained and retained until license termination.

- Records of waste management for any "Naturally occurring and/or Accelerator-produced Radioactive Material" (NARM), specifically, radioactive materials that are naturally occurring and are not source, special nuclear, or byproduct materials. These records will include:
    - descriptions of any radiological spills, excursions, contamination events or unusual occurrences that have the potential to exceed site cleanup standards or that leave the site (These descriptions will address: the dates, locations, areas, or facilities affected; assessments of hazards;

corrective and cleanup actions taken; assessment of cleanup effectiveness; the location of any remaining contamination; nuclides involved; quantities, forms, concentrations, and descriptions of hazardous constituents; descriptions of inaccessible areas that cannot be cleaned up; and sketches, diagrams, or drawings marked to show areas of contamination and places where measurements were made); and
    - information related to site characterization; residual soil contamination levels; on-site locations used for burial of radioactive materials; hydrology and geology, with an emphasis on conditions that could contribute to groundwater or surface-water contamination; and locations of surface impoundments, Storage Ponds, lagoons, and mine unit aquifer anomalies;
- Records of the results of measurements and calculations used to evaluate the release of radioactive effluents to the environment, including those used to evaluate the release of radioactive effluents to the environment required under the standards for protection against radiation in effect prior to January 1, 1994.
- Records containing information important to decommissioning and reclamation, including:
    - as-built drawings or photographs of structures, equipment, restricted areas, mine units, areas where radioactive materials are stored, and any modifications showing the locations of these structures and systems through time;
    - drawings of areas of possible inaccessible contamination, including features such as buried pipes or pipelines; and
    - pre-operational background radiation levels at and near the site.
- Records of disposal of NARM low-level radioactive waste at a licensed low-level radioactive waste disposal facility (LLRWDF) disposal facility, in accordance with Title 10 of the CFR, Part 61 or NRC Agreement State equivalent regulations; and
- Records of management and disposal actions for the UIC Class I wells, based on WDEQ requirements.

Survey and calibration records will be kept for a minimum of three years and will reflect updated 10 CFR 20 requirements. Records will be kept, until license termination, to demonstrate compliance and to help in evaluating dose, intake, and releases to the environment.

Reporting

Table 5.2-1 summarizes the internal and external reporting required by NRC regulations and company policy. The table will be updated by the RSO as necessary to provide a reliable reference for the radiation safety staff.

Quality Assurance Program

A complete Quality Assurance Program will be written and installed before initiation of construction activity and will address environmental monitoring, effluent monitoring, and health physics. The Table of Contents and Executive Summary for the Quality Assurance Program are included in **Attachment 5.2-1**. LC ISR, LLC will submit a decommissioning plan, which will address quality assurance issues, to NRC for review and approval at least 12 months before initiating decommissioning.

5.2.2 Safety and Environmental Review Panel (SERP)

LC ISR, LLC is requesting a Performance Based License (PBL) condition as part of the license consistent with Section 2.7 of NUREG/CR-6733. If a PBL license condition is approved, LC ISR, LLC will develop a Safety and Environmental Review Panel (SERP) to review proposed changes, tests, or experiments to determine whether they require a license amendment. Changes, tests, or experiments may be conducted without prior NRC approval if:

- they do not conflict with any requirements specifically stated in the license or impair the licensee's ability to meet all applicable NRC regulations;
- there is no degradation in the essential safety or environmental commitments in the license application or those provided in an approved reclamation plan; and
- they are consistent with NRC conclusions regarding actions analyzed and selected in the facility environmental assessment.

A license amendment and/or NRC approval must be sought and received before implementation if the proposed change, test, or experiment would:

- result in any appreciable increase in the frequency of occurrence of an accident previously evaluated in the license application (as updated);
- result in any appreciable increase in the likelihood of occurrence of a malfunction of a structure, system, or component (SSC) important to safety previously evaluated in the license application (as updated);
- result in any appreciable increase in the consequences of an accident previously evaluated in the license application (as updated);
- result in any appreciable increase in the consequences of a malfunction of an SSC previously evaluated in the license application (as updated);
- create a possibility for an accident of a different type than any previously evaluated in the license application (as updated);
- create a possibility for a malfunction of an SSC with a different result than previously evaluated in the license application (as updated); or

- result in a departure from the method of evaluation described in the license application (as updated) used in establishing the final safety evaluation report (FSER), or the environmental assessment (EA) or TERs or other analysis and evaluations for license amendments.

5.2.2.1 Organization of the SERP

The SERP will consist of at least three individuals. One member will have expertise in management and will be responsible for implementing managerial and financial changes. One member will have expertise in operations and/or construction and will have responsibility for implementing any operational changes. One member will be the RSO, or equivalent, with the responsibility for assuring that changes conform to radiation safety and environmental requirements. Additional members may be included in the SERP, as appropriate, to address specific technical issues such as health physics, groundwater hydrology, surface-water hydrology, or other technical disciplines. Temporary members may include consultants. Additional members may be included in the SERP when the magnitude of the Project or technical complexity merit.

5.2.2.2 SERP Responsibilities

This procedure will be used for the evaluation of all major changes to the Project's operations. The changes may be derived from operational, economic, or regulatory requirements. The following reviews shall be carried out by the SERP:

Operations/Technical Review

- Review operating criteria and critical equipment and determine the following.
    - Does the proposed change impact the operations as described in the license application?
    - Does the proposed change significantly change the processes used at the facility as described in the license application?
- Review the SOP for the proposed change and determine the impact on existing SOPs. Make the necessary changes to the existing SOPs.
- If applicable, review the emergency response plan and determine compatibility with the proposed change.

Environmental / Health Physics / Safety Review

- Review the proposed change to determine if any changes in monitoring and record keeping are required to ensure compliance with existing programs.

- Review the proposed changes and determine the need for additional training.
- Review key personnel training records and determine training needs as required by the proposed change.

Compliance Review

- Review the proposed change and determine whether it will conflict with Project policies regarding training and safety.
- Review the proposed change and determine compliance with the Project license.
- Review the proposed change and determine compliance with NRC regulations and other federal and state regulations.
- Review the proposed change to determine if any adjustment to the financial surety would be necessary. If the surety will increase as a result of implementing the proposed action, the surety must be updated through a license amendment or the annual surety update before the proposed change takes place.

Based upon the criteria listed in **Section 5.2.1**, the SERP will determine whether to seek a license amendment for the proposed change, test or experiment or proceed with implementation.

5.2.2.3 Record Keeping and Reporting

Detailed records of all SERP proceedings and findings will be maintained until license termination. The records will include:

- a description of the proposed change, test or experiment;
- the names and titles of each SERP member;
- the findings of each point outlined in **Section 5.2.1.2**; and
- conclusions and recommendations of the SERP including required actions, deadlines, and assignment of responsibility.

These records shall be maintained by the RSO with copies distributed to the Mine Manager and the Manager of EHS and Regulatory Affairs. An annual report summarizing all SERP actions will be submitted to the NRC.

5.3 Management Audit and Inspection Program

A variety of inspections and audits will be performed at defined intervals in order to ensure continual protection of employees, the public, environment, and operation. The inspections and audits are to be as follows.

5.3.1 Radiation Safety Inspections

5.3.1.1 Daily Inspections

The RSO, HPT, or an individual designated by the RSO (hereafter referred to as the Designee) will conduct a daily inspection of all Plant areas where radioactive materials are present and/or where direct radiation levels may be elevated. The areas inspected will include, but shall not be limited to, the Plant, byproduct storage area, and Storage Ponds. The inspector will look for and report to the Operations Manager, Site Supervisor EHS/RSO and Mine Manager all non-conformances with regulations, SOPs, and ALARA principle. The inspector shall record the date, his name, areas inspected, and findings for each inspection on the inspection checklist. If corrective actions are necessary, they shall be implemented as soon as is practicable. Corrective actions taken shall be documented. Documentation shall be maintained until license termination. At a minimum, the inspector will specifically check the ventilation systems, signage, security features, and the status of the Continuous Working Level (CWL) monitoring device.

The RSO may only designate an individual to perform daily inspections if that individual meets the training and experience qualifications outlined in Sections 5.4.3.2 and 5.5. A Designee may only be appointed to perform daily inspections that occur on weekends, holidays, and times when both the RSO and HPT(s) must both be gone at the same time (for example, illness or offsite training). In no case shall a Designee perform daily inspections for more than three consecutive days. The Designee has no authority to perform health physics duties outside the scope of his/her regularly assigned duties. For example, the Designee will not have authority to release materials for unrestricted release or to approve a RWP. On the first day the RSO or HPT return to work, the daily inspection checklist used by the Designee must be reviewed by the RSO or HPT. During periods when a Designee is used to complete daily inspections, either the RSO or HPT must be reachable by telephone to provide assistance.

5.3.1.2 Weekly Operations Inspections

The RSO and Operations Manager, or their designees in their absence, shall perform a weekly inspection of all areas of the facility where radioactive materials and/or radiation levels above background may be present. The inspectors shall ensure that all regulations, SOPs and ALARA principles are being followed. The inspectors shall also look for ways to improve the operation in order to minimize exposure to radionuclides. The RSO or their designee shall perform the weekly inspection by listing the date, areas visited, names of inspectors, and inspection findings. Inspection findings shall be reported to the

Mine Manager, RSO, and Manager of EHS and Regulatory Affairs. Documentation shall be maintained until license termination.

5.3.1.3 Monthly RSO Reports

Pursuant to NRC Regulatory Guide (RG) 8.31, at least monthly, the RSO will review the results of daily and weekly inspections, including a review of all monitoring and exposure data for the month. The RSO will provide the Mine Manager and Department Heads a written summary of the month's significant worker protection activities that contains a summary of the most recent personnel exposure data, including bioassays and time-weighted calculations and a summary of all pertinent radiation survey records. In addition, the monthly summary report should specifically address any trends or deviations from the radiation protection and ALARA program, including an evaluation of the adequacy of the implementation of license conditions regarding protection and ALARA. The summary should provide a description of unresolved problems and the proposed corrective measures. Monthly summary reports will be maintained on file and be readily accessible for at least five years.

5.3.2 Storage Pond Inspections

Storage Ponds will be installed at the facility to act as surge capacity. The Storage Ponds will be constructed and inspected in accordance to applicable guidance found in NRC RG 3.11.1. In the event of a significant occurrence such as a flood, tornado, earthquake, or intense rain, the Manager of EHS and Regulatory Affairs may have additional Special Inspections performed to ensure the continued stability of the Storage Ponds.

5.3.2.1 Daily Storage Pond Inspections

The following inspection will be performed by a trained employee with the results documented on an official form. Documentation will be maintained by the Site Supervisor EHS/RSO until license termination. The Operations Manager and Manager of EHS and Regulatory Affairs shall review the results of the daily inspection.

- The condition of inlet and outlet piping and associated valving shall be visually inspected to ensure they are correctly positioned and in good operating condition with no obvious damage.
- Water levels shall be recorded and referenced against allowable freeboard to ensure safe levels are maintained.
- The retention dam and diversion ditches shall be visually inspected for signs of cracking, movement, erosion and seepage.

- When in use, the enhanced evaporation system shall be inspected daily for proper operation.
- An automated leak detection system will be installed in the leak detection standpipe. If the system detects a leak it will notify the operators with an alarm.

5.3.2.2 Weekly Storage Pond Inspections

The following inspection will be performed by a trained employee with the results documented on an official form. Documentation will be maintained by the Site Supervisor EHS/RSO until license termination. The Operations Manager and Manager of EHS and Regulatory Affairs shall review the results of the weekly inspection.

- The perimeter fence and associated signage shall be inspected to ensure adequate protection from wildlife intrusion and warning of potential hazards, respectively.
- The automated leak detection system will be inspected to ensure it is working properly.
- Diversion channels shall be inspected for erosion.
- Emergency ropes shall be inspected for proper placement and good working condition.
- The pond liner shall be inspected for holes and signs of stress.

5.3.2.3 Quarterly Storage Pond Inspections

Since the shallowest aquifer is isolated from the storage ponds by aquitards, four monitor wells will be completed immediately above the shallowest aquitard down-gradient of the storage ponds. At least quarterly, water level readings will be taken to see if any water is collecting at the surface of the aquitard. If water is present, an attempt will be made to collect a sample that will be analyzed for specific conductance, chloride, alkalinity, sodium, and sulphate. If the water chemistry in the well is similar to pond water chemistry, an investigation will be made to determine if a pond is leaking.

The following inspection will be performed by the Site Supervisor EHS/RSO with the results documented on an official form. Documentation will be maintained by the Site Supervisor EHS/RSO until license termination. The Operations Manager and Manager of EHS and Regulatory Affairs shall review the results of the weekly inspection.

- The top of the embankment and toe areas shall be examined for evidence of settlement, seepage, erosion, or depression.
- Water quality results from the leak detection standpipes and groundwater monitor wells will be reviewed for evidence of leakage.

- Embankments will be inspected for cracks, movement, irregularities in alignment, and erosion.

5.3.2.4 Annual Technical Evaluation of Storage Ponds

The following inspection will be performed by the Manager of EHS and Regulatory Affairs, who may elect to receive assistance from outside technical experts. Documentation of the inspection findings and potential corrective actions will be maintained by the Site Supervisor EHS/RSO until license termination. The Operations Manager and Manager of EHS and Regulatory Affairs shall review the results of the Annual Technical Evaluation and ensure all necessary corrective actions are completed.

- The findings from the previous year's daily, weekly, and quarterly inspections shall be reviewed to ensure they are thorough, properly documented, and that findings have been appropriately corrected.
- An assessment of the hydraulic and hydrologic capacities shall be made to ensure the proper infrastructure is in place.
- The embankment shall be surveyed to ensure movement is within acceptable ranges.
- The inspector shall visually inspect the embankments, embankment toes, and diversion ditched to ensure there is no seepage, undesirable movement, or erosion.
- The water quality of the Storage Ponds shall be determined and compared against any trends in ground and surface water quality.

5.3.3 ALARA Operating Philosophy

LC ISR, LLC commits to abiding by the principle of As Low As Reasonably Achievable (ALARA) as outlined in NRC RG 8.10. The ALARA philosophy involves a formal and binding commitment by management to provide clearly defined radiation protection responsibilities and an environment in which the radiation protection staff can do its job properly. It will be the responsibility of the RSO and radiation protection staff to conduct surveillance programs and investigations to ensure that occupational exposures are as far below the specified limits as is reasonably achievable. Additionally, the RSO and radiation protection staff should be vigilant in searching out new and better ways to perform all jobs with lower radiation doses. The RSO is assigned sufficient authority to enforce safe ALARA operations and employees are trained to understand and apply the ALARA philosophy.

As part of implementing the ALARA Philosophy, an annual audit of the radiation safety and ALARA programs will be performed by the Manager of EHS and Regulatory

Affairs, General Manager, and the Operations Manager. The Manager of EHS and Regulatory Affairs may also call on outside technical expertise to complete the audit. A technical expert for the purposes of this section shall be an individual who meets the qualifications of an RSO and who has at least five years of experience in applied radiation safety. The Site Supervisor EHS/RSO may be called upon to provide data but shall not be involved in audit findings or the writing of the Annual ALARA Audit Report.

The purpose of the audit shall be to: 1) determine the effectiveness of the radiation safety and ALARA programs and ensure the veracity of radiation measurements and calculations; 2) ensure compliance with applicable regulations, procedures, and policies; 3) ascertain trends in employee and public exposure and potential reasons for trends; and 4) look for methods to further mitigate employee and public exposure to radionuclides. The Annual ALARA Audit shall be conducted in accordance with NRC Regulatory Guide 8.31. A written report of the audit findings will be submitted to the President, General Manager, Mine Manager, and all Department Heads. Additionally, the report findings and their implications shall be discussed with all employees during annual radiation safety training.

The Annual ALARA Audit Report shall summarize:

- employee exposure records (external and time-weighted calculations);
- bioassay results;
- inspection log entries and summary reports of daily, weekly, and monthly inspections;
- documented training program activities;
- radiation safety meeting reports;
- radiological survey and sampling data;
- reports on overexposure of workers submitted to the NRC and other applicable regulatory agencies; and
- operating procedures that were reviewed during this time period.

The report shall specifically address the following:

- trends in personnel exposures for identifiable categories of workers and types of operational activities;
- whether equipment for exposure control is being properly used, maintained, and inspected; and
- recommendations on ways to further reduce personnel exposures from uranium and its daughters.

5.4 Qualifications for Personnel Conducting Radiation Safety Program

The minimum qualifications and experience levels required of personnel assigned the responsibility of developing, conducting, and administering the Radiation Safety Program are described below.

5.4.1 Mine Manager

The position of Mine Manager requires a bachelor's degree in engineering or associated science from an accredited college or university, plus a minimum of five years of managerial experience and directing operational functions.

5.4.2 Manager of EHS and Regulatory Affairs

This position requires a bachelor's degree in an engineering or science field degree from an accredited college or university, or an equivalent level of work experience. Additionally, a minimum of five years in senior management and operations functions will be required as well as the ability to meet the requirements of the position of RSO.

5.4.3 Site Supervisor EHS / RSO

The Site Supervisor EHS/RSO must demonstrate a working knowledge and proper understanding of the operation of radiation health physics instruments and equipment used during uranium recovery, surveying and sampling techniques, and personnel dosimetry requirements. In accordance with NRC RG 8.31, the position of RSO requires:

- a bachelor's degree in physical science, industrial hygiene, or engineering from an accredited college or university or an equivalent combination of training and relevant experience in radiation protection related to uranium recovery (Two years of relevant experience are generally considered equivalent to one year of academic study.);
- at least one year of work experience relevant to uranium recovery operations in applied health physics, radiation protection, industrial hygiene, or similar work (This experience should involve actually working with radiation detection and measurement equipment, not strictly administrative or "desk" work.);
- at least four weeks of specialized classroom training in health physics specifically applicable to uranium recovery (In addition, the RSO should attend refresher training on uranium recovery health physics every two years.); and

- a thorough knowledge of the proper application and use of all health physics equipment used during uranium recovery activities, the chemical and analytical procedures used for radiological sampling and monitoring, methodologies used to calculate exposure to uranium and its daughters, and a thorough understanding of the uranium recovery process and equipment used and how the hazards are generated and controlled during the uranium recovery process.

5.4.3.1 Health Physics Technician

The HPT will have one of the following combinations of education, training, and experience.

Option I:

- an associate degree or two or more years of study in the physical sciences, engineering, or health related field;
- at least a total of four weeks of generalized training (up to two weeks may be on-the-job training) in radiation health protection applicable to uranium recovery facilities; and
- one year of work experience using sampling and analytical laboratory procedures that involve health physics, industrial hygiene, or industrial safety measures to be applied in a uranium recovery facility; or

Option II:

- a high school diploma;
- a total of at least three months of specialized training (up to one month may be on-the-job training) in radiation health protection relevant to uranium recovery facilities; and
- two years of relevant work experience in applied radiation protection.

5.4.3.2 Designee

A "Designee" for the purposes of inspections shall have the following qualifications: a minimum of a high school diploma; training as a radiation worker pursuant to TR Section 5.5 and RG 8.31, Section 2.5; on the job training on how to use the daily inspection checklist, and at least three months experience working as a radiation worker at the Lost Creek Facility.

The Designee has no authority to perform health physics duties outside the scope of his/her regularly assigned duties. For example, the Designee will not have authority to release materials for unrestricted use or to approve a RWP.

5.4.4 Department Heads

These positions require a bachelor's degree in engineering or associated science degree from an accredited college or university or an equivalent level of work experience, plus a minimum of two years of managerial experience in engineering, geology, or operational functions.

5.5 Radiation Safety Training

Employee training will be designed to familiarize employees with all of the necessary precautions to be taken when performing their assigned duties, and radiation safety constitutes a significant portion of this training. New and experienced employees alike will be provided written copies of radiological safety instructions, and will take training courses that address the fundamentals of radiation protection and potential risks of radiation exposure.

The radiological protection program for LC ISR, LLC uranium recovery operations will, in general, include annual worker training to insure that site personnel will, at all times, have sufficient awareness and continuity of knowledge regarding:

- general safety regulations, principals, and procedures;
- the fundamentals of health protection, personal hygiene, and housekeeping requirements;
- basic radiation science and radiation safety principals;
- the Radiation Safety Program for ISR operations at the Permit Area, including all site-specific and operation-specific radiation safety procedures, and radiation protection regulations;
- dose monitoring requirements and procedures and health protection measurements;
- worker rights, responsibilities and notifications, and facility-provided protection;
- contamination and spill control; and
- security and emergency procedures.

Additionally, radiation safety training for female employees will address:

- risks associated with prenatal radiation exposure, and
- the LC ISR, LLC policy for declared pregnant women, including dose limits and rates.

Managers will also receive additional specialized occupational radiation protection training on their supervisory responsibilities. Each permanent employee that has

completed the new employee radiation safety training will annually attend an abbreviated retraining course.

The refresher course will discuss:

- relevant information that has become available during the past year;
- a review of safety problems that have arisen during the past year including results from the ALARA report;
- changes in regulations and license conditions;
- exposure trends; and
- other current topics.

A written or oral test will be conducted following radiation safety training for new employees and annual refreshers. Incorrect answers to test questions will be discussed to ensure a correct understanding of the material. If an employee fails to pass the test (less than 70 percent of the answers being correct), additional training will be provided prior to re-testing. Tests and results will be maintained on file until license termination.

Continual training will be conducted to ensure that personnel maintain awareness of events and issues that could affect the quality of program performance. At least quarterly, employees will be updated on radiation safety issues that arise during the Project.

Specific, detailed worker radiation training materials will be presented in the Radiation Safety Manual which will include materials for initial employee training (eight hours) as well as for ongoing refresher training (four hours), which will occur on an annual basis for each employee. The RSO and HPT will complete 40 hours of appropriate radiation safety refresher training by qualified instructors on a biennial basis. Training of all personnel will be documented with records maintained by the RSO until the license is terminated.

Visitors and contractors will be required to sign in at the office in the Plant and receive appropriate hazard recognition and safety training. Visitors will be instructed on radiological and non-radiological hazard prevention specific to the areas of visitation. Contractors who handle contaminated equipment will receive the same training and radiation safety instruction required of permanent employees. Contractors, who have previously completed the full training for the Project or who have evidence of recent and relevant training elsewhere, will receive job-specific radiation safety instruction. All visitors and contractors that have not received proper training must be escorted by an employee with proper training and knowledge of potential hazards.

5.6 Permit Area Security

LC ISR, LLC is committed to:

- maintaining control of NRC licensed material;
- providing a safe and secure workplace;
- managing records that may contain sensitive and/or confidential information; and
- ensuring safe and secure transportation of NRC licensed material in accordance with applicable DOT and NRC regulations and guidance.

5.6.1 Mine Unit and Storage Pond Security

Each mine unit and Storage Pond will be fenced and have signage to prevent inadvertent entry by members of the public. Additionally, during production, active mine units will be inspected by operations personnel at least once per shift. Employees are instructed to inform their supervisor if any unauthorized individual gains access to the mine units. The supervisor will request the intruder to leave. If the intruder refuses to leave, the supervisor will request assistance from the County Sheriff and notify the Mine Manager. Employees shall not confront trespassers if they feel their safety may be in jeopardy.

Visitors to the mine units will be required to register at the office so appropriate training can be completed and any necessary supervision assigned. Visitors will only be accepted during normal business hours unless approved by a member of management.

5.6.2 Plant Security

The Plant, including areas of byproduct storage and handling, shall be fenced with access controlled by a locked gate. This controlled access area is designated in this Technical Report as the "controlled area". Signage will warn employees and members of the public of the potential for exposure to radionuclides and the necessity for members of the public to register at the office before entering. Security cameras will be placed at strategic locations throughout the Plant, including the security gate and locations where source and byproduct materials are stored.

The Plant will normally operate 24 hours per day, seven days per week. During operations, employees will be continuously on-site to monitor security cameras and account for licensed material. At the beginning of each shift, the Plant Operator will account for the quantity and location of all source material in storage. The Plant Operator will document his/her findings so the next shift can make an accurate account.

Employees are instructed to inform their supervisor if any unauthorized individual gains access to the Plant. The supervisor will request the intruder to leave. If the intruder refuses to leave, the supervisor will request assistance from the County Sheriff and notify the Mine Manager. Employees shall not confront trespassers if they feel their safety may be in jeopardy.

Visitors to the Plant will be required to register at the office so appropriate training can be completed and any necessary supervision assigned. Visitors will only be accepted during normal business hours unless approved by a member of management.

The Operations Manager shall minimize the quantity of byproduct material stored at the site in order to minimize any potential security threat.

5.6.3 Transportation Security

Licensed material is most vulnerable to security threats while it is being transported. Therefore, LC ISR, LLC commits to the following practices involving the transportation of licensed material.

- All individuals involved in the packaging, labeling, and handling of licensed material will be trained in applicable DOT regulations, including the Security Plan as well as the facilities radiation safety SOPs and policies.
- Licensed material will be packaged, labeled, placarded and adequately described in shipping papers in accordance with applicable DOT and NRC regulations.
- Shipments of licensed material shall be kept within the controlled area of the Plant and shall remain locked prior to shipping.
- Bulk shipments of licensed material shall be secured by locking trailers and vehicles when they are not occupied.
- Bulk shipments of licensed material will only be sent on exclusive use carriers.
- All drivers transporting bulk quantities of licensed material will be familiar with the hazards of the shipment and how to properly respond to accidents involving the material.

5.7 Radiation Safety Controls and Monitoring

5.7.1 Effluent Control Techniques

During the Project, gaseous/airborne, liquid, and solid effluents will be produced from the processes associated with ISR operations. The only gaseous emission of concern due

to radiation is radon gas, which will be vented to the atmosphere (**Sections 4.1.2 and 5.7.1.1**. No yellowcake drying and packaging will occur within the Permit Area, and the Storage Ponds will be kept wet, reducing the potential for radioactive particulate emissions as compared to other ISR projects in Wyoming. Since there will be no source of airborne particulate that is of concern outside the Plant, monitoring for airborne particulate effluent at the Plant boundary will not be necessary. A more detailed explanation of the risks from airborne particulates can be found in **Section 5.7.1.2**.

The three liquid 11(e)(2) byproduct materials generated will be: process wastes; groundwater affected by ISR and pumped out for purposes such as sample collection; and groundwater generated during aquifer restoration. These liquids will be treated and disposed of on-site through a system of Storage Ponds (**Section 5.7.8.3**) and UIC Class I wells (**Section 5.7.8.2**). The solid 11(e)(2) byproduct materials will include process wastes, such as spent ion exchange resin, and will include equipment that becomes contaminated during ISR operations and ore processing (**Section 4.3.2**). Where possible, equipment will be decontaminated for disposal as non-11(e)(2) material or for re-use. Equipment that cannot be decontaminated and process wastes will be disposed of at an NRC-licensed disposal facility. Detailed information about all of the effluents generated during ISR operations, including those described above that are of additional concern due to radiation, is provided in **Section 4.0**.

5.7.1.1 Release of Radon

Due to the remoteness of the facility, the administrative and engineering controls to be implemented, and the nature of ISR, the potential for a member of the public to be exposed to radon levels exceeding the limits outlined in 10 CFR 20 Appendix B is minimal. Radon is the only effluent which will routinely leave the controlled site in a manner that could result in a radiation dose to a member of the public. MILDOS modeling, in **Attachment 7.2-1** of this report, demonstrates that the possible dose at the nearest downwind Permit Area boundary from the Plant due to radon is three mrem per year while the regulatory limit is 100 mrem/year. Operational monitoring for radon is described in **Section 5.7.3.3**.

NUREG/CR-6733 considers the risk involved with the spill of a large quantity of lixiviant and the resulting release of radon. The NUREG demonstrates that even if the spill event is uncorrected for eight hours, the exposure to a member of the public at the boundary will be below regulatory limits. However, the dose to an employee may exceed regulatory limits if the spill is not handled properly. In order to ensure the dose limit is not exceeded, the engineering and administrative controls found in **Section 5.0** will be implemented.

Radon effluent will be controlled by utilizing pressurized ion exchange columns and ventilation. Each unpressurized vessel that may be a source of radon gas will be vented outside the Plant in a manner that minimizes exposure to employees and the public (**Figure 4.1-1**). A general area ventilation system will also be installed to provide fresh air to the work area in the event radon is released in concentrations of concern.

There are three main areas the Plant, header houses, and mine units, where radon effluent will be released and ventilated accordingly. These areas are discussed below.

Plant

The Plant will utilize three modes of ventilation to control radon effluent.

- Passive Ventilation: incorporates natural ventilation within the Plant including doors and overhead doors to reduce radon levels within the Plant proper. An example of this type of ventilation is the opening of the overhead doors during a resin transfer to allow natural ventilation or cross flow within the building.
- Area Ventilation: incorporates wall or area fans to ensure an adequate exchange of air in areas where radon may be more likely to gather or in which Plant air is otherwise stagnant. The fans will be placed in such a manner as to move air closer to the ground where radon is expected to accumulate. Routine monitoring and sampling will be used to verify the effectiveness of the fan placement and volume of air moved. Fan discharge will be either directly outside the building if the fan is mounted on an exterior wall or ducted outside in the case of an interior wall.
- Point Source Ventilation: incorporates direct ventilation to the roof or outside of the Plant utilizing a stack and, in some cases, a fan to increase the ventilation rate. An example of this type of ventilation is the ion exchange columns when they are de-pressurized to allow for transfer of resin. The ventilation will go from the column to a knockout pot (to insure no particulate discharge) then outside the building. Another example of point source ventilation is the venting of sumps. Each sump will have a stack with an inline fan to pull the radon gas from the sump and discharge it through the roof of the building.

Header Houses

The header houses will utilize two styles of ventilation to control radon effluent.

- Passive Ventilation: incorporates natural ventilation within the building including doors to reduce radon levels. An example of this type of ventilation is the opening of the doors during header house checks and sampling to allow natural ventilation or cross flow within the building.

- Area Ventilation: header house forced ventilation incorporates a wall fan, with the suction located in the basement of the header house. The wall fan captures air from the basement and discharges it out the rear wall of the header house. This fan is designed to operate 24 hours a day, year round.

Mine Units

The mine units will utilize passive ventilation to control radon effluent at the wellheads. Each injection and production wellhead is a potential release point for radon. Each wellhead will be covered with a loosely fitting, insulated cover. Prior to inspecting or working on a well, the lid to the wellhead cover will be removed and allowed to ventilate for a short period of time. Any radon that may have built up in the cover will quickly dissipate at that time.

Ventilation System Maintenance

Testing, maintenance, and inspection of the ventilation equipment will be done to the manufactures recommendations. Although the final equipment has yet to be specified or ordered, the required size and flow rates are known. Using representative systems as references, periodic maintenance will be required on the belts, bearings, fasteners, including lubrication and removal of dirt. Since the representative manufactures do not provide more descriptive maintenance schedules, LC ISR, LLC initial maintenance schedule will be to check all of the ventilation equipment quarterly. This schedule may be adjusted once operations start and maintenance personnel have evaluated the system's performance.

Section 4.1 provides additional discussion of how the Plant and mine units will be designed to minimize radon effluent. A Continuous Working Level (CWL) radon monitor will be installed in the Plant to warn employees if the level of radon exceeds 25 percent of the derived air concentration (DAC) (0.08 Working Level [WL]). The minimum detectable radon decay product concentration for the Alpha NUCLEAR PRISM 590 Series (or equivalent) CWL is 0.004 WL. The PRISM draws ambient air through a filter membrane and counts the alpha particle radiation collected on the filter. The software program calculates the average WL, maximum WL, and produces periodic WL listing, and graphical WL plots. The device has a three-level LED lamp warning system (green, yellow, and red). The user sets the three warning levels. Employees will be trained to respond appropriately to the warning levels. The CWL will be used only as an alarm device. Radon decay product measurements using the Kusnetz method will be used for estimating worker doses as described in **Section 5.7.3.2**.

5.7.1.2 Release of Airborne Particulate

Accident scenarios resulting in the exposure of a member of the public to concentrations of radionuclides in airborne particulate matter in concentrations above dose limits established in 10 CFR 20.1301 are highly improbable given the remoteness of the facility and the engineering and administrative controls that will be employed. This conclusion is supported by NUREG/CR-6733, which discusses the potential sources of effluent resulting from accidents at an in situ facility. The NUREG considers several accident scenarios including the failure of a thickener. The LC ISR, LLC facility will not have a thickener but will have tanks containing bulk quantities of yellowcake slurry. The authors of the NUREG, using extremely conservative assumptions, found that the only way the dose to a member of the public from this accident scenario could occur is if at least 20 percent of the content of the thickener escaped the Plant and a member of the public was less than one hundred meters from the accident. The nearest boundary to the Plant will be 333 meters away. The scenario also assumes that there will be no timely accident response to recover the spill and the slurry is allowed to dry. Given that the slurry storage tanks will be positioned a considerable distance from exterior walls and within a bermed area it is highly unlikely that a ruptured vessel of yellowcake slurry could reach the outdoor environment. Further, LC ISR, LLC will have procedures for addressing such events and will also train employees how to respond in a timely manner.

NUREG/CR-6733 also considers the risk of accidents involving a yellowcake dryer. Since LC ISR, LLC will not have a yellowcake dryer, these scenarios are not applicable. Section 2.8.1 of NUREG/CR-6733 considers the yellowcake dryer and packaging system the normal source of airborne particulate.

5.7.1.3 Release of Liquid Effluent (Pregnant Lixiviant)

NUREG/CR-6733 considers two conservative scenarios involving the release of pregnant lixiviant and loaded resin. In both scenarios, the authors determined that the spills would have no significant external radiological risks. The risks from associated radon releases are discussed in **Section 5.7.1.1**. All process and effluent liquids will be contained within pipelines, tanks and storage ponds that are inaccessible to members of the public.

Risk Analysis of Pregnant Lixiviant Release

NUREG/CR-6733 does state that these types of spills cannot be discounted from risk assessment on the basis of probability and concludes that effluent levels and internal doses can be significant due to spills of pregnant lixiviant. Therefore, the risk regarding the release of pregnant lixiviant has been analyzed. The site-specific analysis includes:

risk tables, which evaluate likelihood of occurrence along with mitigation practices; spill response; remediation; and exposure pathways.

Risk Tables

Attachment 5.7-1 is an assessment of the operational risks associated with a spill of pregnant lixiviant. The assessment consists of four parts. The first part is a description of the three spill scenarios that were evaluated (e.g., leak in the production line in a header house). The second, third, and fourth parts provide detail about each of the three scenarios, respectively. That detail includes a chart of the likelihood of events occurring in a given scenario compared to the severity of the consequences, from a radiological standpoint, should those events occur. The detail also includes discussion of the risks associated with the events and the operational procedures to help prevent those events and mitigate the consequences (e.g., sump alarms in the header houses).

Spill Response and Remediation

Section 4.2.5.5 of this Technical Report addresses prevention and remediation of spills, including identification of those facilities which require particular attention, such as pipelines.

Exposure Pathway Analysis

Both employees and members of the public may be exposed to radioactive constituents in a spill through: direct exposure; inhalation/ingestion of fluids or dry particulates; and/or inhalation of radon. Exposure to radon is not considered a significant concern since it rapidly disperses outdoors.

Before a spill dries, the generation of airborne particulates and airborne fluids is improbable since there is no credible mechanism of generation. Because of the arid conditions, absorption of spilled liquid into the soil is of more concern than potential impact to a drainage.

Once a spill dries it is conceivable, if not likely, that some airborne particulate will be created by the wind. To assess the potential for exposure, it is reasonable to consider Section 4.4.3 of NRC NUREG CR/6733. The spill scenario presented is conservative given the extreme size of the spill (infinite areal extent) and the elevated concentration of Ra-226 (3,400 pCi/L) and uranium (39,700 pCi/L). As discussed in **Section 7.4 and Section 7.6**, there are berms and automatic shut-offs, as well as regular inspections, to limit spill size. Most ISR operations, including the Lost Creek Project, have significantly lower Ra-226 concentrations and slightly lower uranium concentrations. Even with the use of conservative values the analysis reveals that soil impacted by a spill would be

below the decommissioning standard for Ra-226 and uranium. Since this standard is intended to protect a member of the public who may actually live on the spill site after decommissioning, the soil should be considered safe with respect to Ra-226 and uranium. However, in an effort to maintain ALARA, the RSO, upon initiation of operations, shall review the actual radionuclide content of the lixiviant and determine potential doses to employees and members of the public as the result of air particulate derived from credible spill scenarios. The existing SOP will be adjusted, if necessary, to ensure employees safely recover spills and effected soil and minimize the potential for public exposure. The RSO, as required, will review the SOP annually to ensure continued accuracy and relevance.

External exposure to radionuclides in the spill is addressed in NUREG CR/6733 and is characterized as a "negligible hazard" despite using conservative examination. The concentrations of radionuclides used in the NUREG analysis are significantly higher than those anticipated in the lixiviant at Lost Creek but the exposure pathway is the same. However, in an effort to maintain ALARA, the RSO, upon initiation of operations, shall review the actual radionuclide content of the lixiviant and determine potential doses to employees and members of the public as the result of external exposure resulting from credible spill scenarios. The existing Standard Operating Procedure (SOP) will be adjusted, if necessary, to ensure employees safely recover spills and effected soil and minimize the potential for public exposure. The RSO, as required, will review the SOP annually to ensure continued accuracy and relevance.

5.7.1.4 Prevention of Accidental Releases

The prevention of and response to accidents that could result in the release of radioactive materials will be considered in the following ways in order to minimize exposure to employees and members of the public.

- An emergency response manual will be developed and will include actions to be taken to minimize and monitor the exposure to employees and members of the public in the event of an unplanned release.
- Employees will be trained how to respond to plausible emergency scenarios.
- A fire prevention program will be implemented and will address issues such as storage of flammables and combustibles, housekeeping, electrical safety, employee training, emergency response to fires, and hot work.
- SOPs will address upset conditions and the appropriate response to be taken by the employee. When an SOP is not in place to cover radiologic work, an RWP will be developed and implemented.

- Semi-daily safety inspections will be made of the Plant and the chemical storage areas to look for potential safety issues that may result in a release of radioactive materials.
- The Plant will be manned around the clock during operations.
- A fire alarm system will be installed in the Plant and office building.
- Instrumentation will be installed in the Plant and header houses to alert operators of any significant change in operating conditions.
- Access to the Plant will be secure. Members of the public will only be allowed access after they have received site-specific training and are escorted by an authorized employee.
- The Plant will be designed and constructed in compliance with all applicable state electrical and fire code regulations. The state or county will inspect all construction and electrical work to ensure they are compliant with applicable codes.
- All vessels, piping, and associated fittings will be compatible with the chemicals, pressures and ambient conditions expected during operations and plausible upset conditions. The risks associated with interaction of chemicals and radioactive materials are evaluated in the risk assessment table in **Attachment 5.7-2**. The table consists of five parts. The first part is a description of the three scenarios that were evaluated in which a chemical/radiological interaction could occur (e.g., a fire). The second part indicates where within the plant a chemical could possibly react with piping or tank materials, potentially resulting in contact with radioactive materials (e.g., leakage due to pipe corrosion). The second, third, and fourth parts provide detail about each of the three scenarios, respectively. For each scenario, the detail includes a chart of the likelihood of interaction of the chemical with radioactive materials compared to the severity of the consequences of that interaction, from a radiological standpoint, should those events occur. The detail also includes discussion of the risks associated with the interactions and the operational procedures to help prevent those interactions and mitigate the consequences (e.g., selection of appropriate piping material).
- All access points to the mine units and Plant will be posted so members of the public are aware of the potential for radioactive materials to be present.

5.7.2 External Radiation Exposure Monitoring Program

The commitments outlined in this Section are intended to be minimum practices. The RSO shall retain the authority to increase sampling and monitoring frequency and locations as necessary to ensure adequate protection of all workers and the public. The RSO may determine that increased sampling and monitoring may be needed at the beginning of operations to quickly develop an understanding of baseline values and to help anticipate where additional engineering or administrative controls may be required.

The following aspects of the program are described below: survey equipment calibration; external exposure rates, and expected doses.

Radiation Survey Equipment and Calibration Methods

Radiation survey instruments will be calibrated by the manufacturer or a qualified calibration laboratory at the manufacturer's suggested calibration intervals, in any case not to exceed one year. Calibration procedures used by Ludlum Instruments, Inc. are included in **Attachment 5.7-3** Standard calibration at Ludlum Measurements, Inc. includes: "as found" readings; two calibration points per range; National Institute of Standards and Technology (NIST) traceable sources; and compliance with American National Standards Institute (ANSI) N323.

Instrument response using a check source in fixed geometry, and background checks will be performed each day the instrument is used. The results of the checks will be recorded on a standard form to be developed based on the individual instrument to be used. Control charts will be maintained for each instrument. The average of at least 20 measurements will be used to determine the standard deviation of the response. The mean of the measurements and three standard deviations above and below the mean will be indicated on the chart. The instrument will be deemed acceptable for use if the daily check source reading is within three standard deviations of the mean.

Alpha

The counting yield (efficiency) for alpha survey equipment used for counting filters and wipes will be determined at least monthly using a NIST traceable standard source. Alpha survey meters used for detecting surface contamination on skin and clothing will be checked at least weekly to determine whether the instrument is operating within the prescribed limits, i.e., within three standard deviations of the mean value, determined as noted above. The RSO or HPT will review the control charts routinely to look for trends in check source response values. If a trend is noted, the instrument will be taken out of service and evaluated.

Survey instruments will be calibrated according to the standard procedures of the calibration facility. Alpha survey meters will be calibrated using a Th-230 source (principal alpha energy = 4.68 MeV) or as prescribed by the calibration facility.

Beta

The beta survey instrument, a Model 3 survey meter or equivalent with a Model 44-9 GM pancake probe, will be calibrated using a technetium-99 source (maximum beta energy = 0.292 MeV) or a Sr-Y strontium-yttrium source with maximum beta energies of 0.546

MeV and 2.27 MeV, respectively. NRC RG 8.30, Appendix C, discusses a GM probe with an open/close window configuration while LCI proposes to use a pancake probe. Appendix C of NRC RG 8.30 refers to a GM probe with a window that can be open or closed. Appendix C also refers to a one-time determination of the correction factor for the instrument. The correction factor can be determined using a pancake probe and any shield that absorbs beta particles. If beta dose rate measurements are deemed necessary based on the presence of dried or aged yellowcake, the beta correction factor will be calculated as described in Regulatory Guide 8.30 and measurements will be made with a shielded and unshielded pancake probe. The beta dose rate will be calculated as described in NRC RG 8.30.

Gamma

The purpose of gamma radiation exposure rate surveys at an ISR facility such as Lost Creek is to determine whether the dose rate in any area of the Operation is such that a worker could receive a dose in excess of 5 mrem in one hour. Such an area would be designated as a "Radiation Area" and marked with the appropriate signage. In addition, gamma survey readings can be used to determine whether a worker is likely to receive a radiation dose in excess of 500 mrem per year, thus requiring personal dosimetry measurements. A second purpose for the external exposure rate surveys is to evaluate whether radiation doses are being maintained ALARA and to note any unusual exposure rates.

The sodium iodide survey meters will be used to measure direct gamma exposure rates. While the meters are energy dependent, they generally overestimate the actual radiation dose rate at the gamma energies commonly associated with uranium recovery, particularly if they are calibrated against cesium-137. The survey meter measurements are not used to calculate doses to workers. Therefore, measurements made with a sodium iodide detector are inherently conservative. Doses to workers will be determined using Thermoluminescent Dosimeters (TLD) or Optically Stimulated Luminescent (OSL) dosimeters provided by a National Voluntary Laboratory Accreditation Program (NVLAP) certified vendor as required by 10 CFR 20 (i.e., when the potential dose to the worker could exceed 500 mrem per year). External dose rates at the permit boundary and various locations within the Lost Creek Permit Area will be measured using TLD or OSL dosimeters.

Since TLD or OSL dosimeters will be used to assess dose to workers and dose rates at the perimeter of the facility, the use of sodium iodide detectors for measuring exposure rate is acceptable. There is no need to cross-calibrate the detectors against a Pressurized Ion Chamber to obtain the true exposure rate.

Survey instruments will be calibrated according to the standard procedures of the calibration facility. In most cases, calibration facilities calibrate gamma survey meters against a cesium-137 source with a gamma energy of 0.662 million electron volt (MeV).

Expected External Exposure Rates Throughout the Plant and Mine Units.

Occupational radiation dose rates at typical ISR facilities are generally less than 200 mrem per year. The areas where direct radiation doses may be elevated are associated with radium precipitation in pipes, and radium and its decay products accumulating in IX resins and other filters. In some areas the dose rate may be as high as 8 mrem/hour. Previous experience with ISR facilities indicates that external gamma doses may be readily controlled to between 100 and 200 mrem per year (Brown, 2007). The dose rates in the mine units are generally indistinguishable from background (Dr. Tom Johnson, Colorado State University Department of Environmental and Radiological Sciences. Personal Communication. April 2009.)

Maximum Expected Doses to Visitors and Contractors

Prior experience at similar ISR facilities and conventional uranium mills indicates that it is unlikely that escorted visitors or contractors on site for limited periods of time will receive doses in excess of 100 millirem (mrem) per year from site operations. It is not possible to estimate the maximum expected dose to members of the public in the restricted areas and other areas of the site prior to site operations, since the areas where visitors and contractors may go and the time they may spend in these areas have not been determined. Such dose rates will depend on a number of factors, including ALARA reviews. Doses to contractors will be estimated on a task-specific basis, and will be controlled to applicable limits.

LC ISR, LLC will implement a radiation safety program for contractors and visitors. All contractors who work in restricted areas unescorted will be considered radiation workers and will receive radiation worker training. Escorted visitors and contractors will receive training commensurate with their potential exposures and the expected duration on site. The RSO will estimate the potential dose to visitors and contractors based on knowledge of where they are likely to be in the plant and current, measured exposure rates.

Prior to any work on the site by contractors, the RSO will estimate the maximum potential dose based on the expected tasks, previously measured external exposure rates in the areas to be occupied by contractors, and estimated exposure to airborne radionuclides. RWPs will be issued for non-routine work. The potential doses to individuals performing such work will be estimated and noted on the RWPs.

Contractors who have a potential to receive doses in excess of 100 mrem in a year will be designated as radiation workers, will receive radiation worker training and will be subject to the same monitoring requirements as LC ISR, LLC site employees including personal dosimetry and bioassay. Doses to contractors will be tracked and recorded on the NRC Form 5 or equivalent if the doses exceed 500 mrem in a year.

Visitors or other unmonitored individuals will not be allowed in areas where the exposure rate exceeds 2.0 mrem/hr per the requirements of 10 CFR 20.1301(a)(2). Visitors will be on site for only short periods of time, minimizing the possibility of receiving significant radiation doses. Therefore, visitors will not be monitored.

5.7.2.1 Personal External Dosimetry

External doses to workers at the site will be monitored by the implementation of an external dosimetry program. As per NRC regulations found in 10 CFR 20.1502a, any individual exposed to radiation sources and meeting the following conditions will be required to wear a dosimeter that is capable of accurately measuring doses associated with beta and gamma forms of radiation:

- adults likely to receive, in one year from sources external to the body, a dose in excess of ten percent of the limits in § 20.1201(a);
- minors likely to receive, in one year, from radiation sources external to the body, a deep dose equivalent in excess of 0.1 roentgen equivalent in man (rem) (one millisievert [mSv]), a lens dose equivalent in excess of 0.15 rem (1.5 mSv), or a shallow dose equivalent to the skin or to the extremities in excess of 0.5 rem (five mSv);
- declared pregnant women likely to receive during the entire pregnancy, from radiation sources external to the body, a deep dose equivalent in excess of 0.1 rem (one mSv); or
- individuals entering a high or very high radiation area.

The RSO will be responsible for managing the external dosimetry program including determining who will be involved in the program in accordance with NRC RG 8.34. All of the types of dosimeters used at the Permit Area will be approved by the National Voluntary Laboratory Accreditation Program (NVLAP) and have a minimum range of 10 millirem (mrem) to 500 rem with an accuracy of at least plus or minus 15 percent. Dosimeter badges will be worn on the torso, between the waist and neck. When not in use, dosimeter badges and the control supplied by the vendor will be stored in an appropriate, routinely accessible background location away from radioactive source materials or excessive heat as determined by the RSO. Workers will be instructed in the proper use and maintenance of the personal dosimeter.

Badges must be checked out by workers and monitored contractors at the beginning of the shift, and checked back in at the end of the shift. Personnel may only wear the badge assigned to them by the RSO. Dosimeter badges will be exchanged once every three months for new badges.

Lost dosimeters must be reported to the RSO as soon as practicable after the loss is noted. The RSO will issue a substitute badge. The missed dose due to the loss of the dosimeter will be calculated by the RSO by pro-rating the measured dose by the time the dosimeter is worn. The RSO will insert a statement into the employee's dose record with the estimated missed dose.

A permanent radiation dose record for each worker will be maintained by the RSO in a format compliant with NRC RG 8.7. Copies will be provided annually to the authorized worker and upon termination of employment. Results from personnel dosimetry will be used to determine the Deep Dose Equivalent (DDE) for use in determining the Total Effective Dose Equivalent (TEDE). Worker doses shall be maintained below levels listed in NRC regulations found in 10 CFR 20 Subpart C. If a worker receives greater than ten percent of the occupational dose limits in 10 CFR 20 Subpart C, the RSO shall perform an investigation to determine the cause and possible methods for lowering the dose rate in accordance with the ALARA principle. The investigation findings and results of any corrective actions will be documented by the RSO.

Dosimeters will also be maintained on the interior office walls at the two locations shown on **Figure 5.7-1** to ensure gamma rates are within acceptable ranges and to recognize trends. The dosimeters will be the same as those described for personnel monitoring above.

5.7.2.2 Direct Readings for External Exposure

Direct gamma exposure rates will be measured at least semi-annually at all employee work stations and near processing equipment that could be a source of gamma radiation (e.g., tanks and filters). Readings will be taken using a Ludlum Model 2350-1 counter with Model 44-10 2" sodium iodide detector or Ludlum Model 19 or equivalent instrument. If the measured exposure rate exceeds the action level of five mrem per hour the area will be designated as a radiation area and clearly posted as in accordance with NRC regulations found in 10 CFR 20.1902(a). The frequency of gamma surveys in radiation areas will be increased to quarterly. An investigation will be performed by the RSO each time a new area becomes a radiation area to determine the cause of the increased gamma levels. The findings and any corrective actions resulting from the inspection will be documented and records maintained until license termination. In

keeping with the ALARA principle, efforts will be made to reduce the gamma exposure rates in all radiation areas. If it is determined the gamma levels cannot be reduced, then exposures will be minimized by implementing engineering and/or administrative controls. The RSO may remove the radiation area designation after determining that the cause of the elevated gamma readings has been mitigated and gamma levels are demonstrated to be below five mrem per hour.

External gamma surveys will be performed with instrumentation with a low range of at most 100 microRoentgens per hour (μR/hr) full-scale and a high range of at least five milliRoentgens per hour (mR/hr) full scale. Gamma meters will be operated according to manufacturer specifications and shall be checked for proper operation each day of use using a check source in a reproducible geometry. Gamma measurements will be made following guidance contained in NRC RG 8.30. Records of all measurements and calibrations shall be maintained until license termination and in a form compliant with NRC RG 8.7.

As a minimum, gamma surveys will be completed in the following locations (approximate number of tanks in parenthesis):

- each ion exchange vessel (14);
- each elution vessel (two);
- each resin shaker deck (four);
- each yellowcake slurry storage tank (two);
- all four sides of each bank of accessible reverse osmosis tubes (seven);
- each waste water tank (two);
- each precipitation cell (four);
- next to the resin transfer water tank T-201;
- each eluant mixture tank (six); and
- office areas next to the Plant (four).

The proposed number of gamma survey points during operations will be approximately 46 points (**Figure 5.7-1**).

Beta radiation is not expected to be a problem since dried yellowcake will not be stored at the site and storage periods for yellowcake slurry should generally be short. On occasions when yellowcake slurry is stored for extended periods, calculations show that the thickness of the fiberglass tank walls will be sufficient to block most of the beta radiation from the Th-234 and Pa-234m decay products of uranium with 0.19 million electron volts (MeV) and 2.28 MeV maximum beta energies, respectively. Assuming that the tank walls are a minimum of 3/8-inch thick and that the density of the fiberglass is 1.0 grams per cubic centimeter (g/cm^3), the density thickness would be approximately 950 milligrams per square centimeter (mg/cm^2). The range of the maximum energy beta

(2.28 MeV) is 1100 mg/cm^2. Only a very small fraction of the Pa-234m beta particles will penetrate the tank wall. Therefore, the beta dose rate from the slurry tanks is expected to be very low compared to the dose rate from exposed yellowcake.

However, to ensure exposures remain below limits, upon initiation of operations, the RSO shall perform and document an evaluation as outlined in NRC RG 8.30, Section 2.4. If beta surveys are necessary, the RSO shall develop a monitoring program detailing frequency, acceptable equipment, calibration, methodology, and location in accordance with NRC RG 8.30. Appendix C.

Worker doses shall be maintained as far below the levels listed in NRC regulations found in 10 CFR 20 Subpart C as is reasonably achievable.

5.7.3 In-Plant Airborne Radiation Monitoring Program

5.7.3.1 Airborne Uranium Particulate Monitoring

Because the materials in the Plant are in a liquid or slurry form, little or no airborne radionuclides are expected. However, surveys for airborne uranium dust will be performed to:

- demonstrate compliance with the occupational dose limits for workers specified in NRC Regulation 10 CFR 20.1201;
- determine if an area needs to be posted in accordance with NRC Regulations found in 10 CFR 20.1902(d);
- determine whether additional precautionary measures are required to comply with NRC Regulations 10 CFR 20.1701 and 20.1702; and
- determine whether occupational exposures to radioactive materials are being maintained ALARA.

All of the processing at the Plant will occur in the form of water based solutions or wet slurry. Therefore, airborne uranium concentrations should be very low. To ensure airborne levels are below regulatory limits found in 10 CFR Part 20.1201 and Appendix B, monthly air quality samples will be collected at the slurry storage tanks and filter press since these areas have the highest potential for exposure. Sampling will be performed in accordance with NRC RG 8.25. The regulated air sampler, F&J LV-1 or equivalent, will be calibrated according to manufacturer specifications or at least semi-annually with a mass flow meter or primary calibration standard. Samples will be collected on glass fiber filters. Documentation of calibrations and readings shall be maintained by the RSO until license termination and in a form compliant with NRC RG 8.7.

Breathing zone sampling will be implemented as defined in SOPs, at the RSO's discretion when an RWP is used, and anytime a worker is performing a special high-exposure task and may be exposed to more than 12 DAC-hours in any one week. Breathing zone samples will be collected in the vicinity of the user's upper torso so it accurately reflects the air quality being inhaled. The samples will be taken using an MSA Escort ELF pump or equivalent and appropriate filter cassette. Air samplers shall be calibrated as per the manufacturer's recommendation or at least semi-annually. Documentation of calibrations shall be maintained by the RSO.

The RSO shall perform an investigation to determine the cause and develop corrective actions if a worker receives greater than ten percent of the allowable dose in 10 CFR 20 Subpart C.

The quantity of air sampled and the method for analysis should allow a lower limit of detection (LLD) of at least 1 x 10^{-11} microcuries per milliliter (µCi/mL) as per RG 8.30. This LLD is applicable for LCI ISR, LLC's uranium product, solubility class D and W; however TR Section 5.7.3.2 includes a commitment to analyze the alpha emitting nuclide mix, which may require determining alternate LLD. Due to the importance of breathing zone samples in detecting potentially elevated areas of airborne radioactivity, samples will normally be analyzed within two days of sample collection. However, a minimum delay of 24 hours will normally be instituted to allow for the decay of radon daughters. If a situation exists that requires immediate action, the filter can be counted within a few hours to obtain a first estimate of the air concentration with a second count taken after a delay of 24 hours to verify the long-lived alpha concentration. A Ludlum Model 2221 counter and Model 43-1 alpha counter with a Model 180-16 sample holder or equivalent will be used to measure activity on filters. The efficiency of the counting system will be determined each day, prior to use, with a NIST-traceable standard alpha source. A conservative correction factor of 0.5 will be applied to the efficiency to account for sample self-absorption unless more realistic values are found in the literature. Alternatively, the actual sample self-absorption factor can be determined experimentally. The results of area and breathing zone sampling shall be used to calculate employee exposures to airborne radioactivity as described in **Section 5.7.4**.

Any time maintenance is performed, there is a change in operations or when there is an upset condition which creates the possibility of airborne particulate, the RSO will ensure potentially affected employees wear a calibrated breathing zone sampler. Examples of times when breathing zone samples will be collected include but are not limited to: when breaking into a pipe or vessel containing yellowcake slurry or scale resulting from process solutions or eluant, when yellowcake or eluant is spilled, when changing filters contaminated with yellowcake slurry, and when manually handling yellowcake slurry. When issuing an RWP, the RSO or HPT will determine the need for collecting breathing zone samples. The results of sampling will be used to ensure the employee was not

exposed to elevated airborne particulate and to verify that existing procedures are ALARA.

5.7.3.2 Establishing Derived Air Concentrations

Initial establishment of appropriate Derived Air Concentrations (DAC) based on both solubility considerations and on the potential mixture of radionuclides is discussed in the section. The procedures LC ISR, LLC will use to determine site-specific solubility information are also discussed.

Solubility Considerations

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. Uranium will be present at the facility exclusively in relatively soluble forms, i.e., uranyl carbonates (various forms), uranyl trioxide (UO_3), uranyl peroxide (UO_4), and their hydrates. 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. However, when acid is added in the precipitation circuit (TR Section 3.3.3), the carbonate complexes dissociate to form uranyl ions. When hydrogen peroxide is added in the precipitation circuit, the uranium is oxidized further to form uranyl peroxide (UO_4*nH_2O). If in the future LC ISR, LLC installs a vacuum drier operating at relatively low temperature, a combination of the soluble forms UO_4, UO_3 and their hydrates will be the uranium compounds of interest throughout the Plant. Nonetheless, initially LC ISR, LLC will assume all uranium compounds to which workers may be exposed are solubility Class W per the International Commission on Radiological Protection [ICRP] Publication 19), which is equivalent to Class M (ICRP 66), until such time as solubility characteristics can be further assessed. The basis for the assumption that the uranium compounds are solubility Class W is described below.

Although specific studies and references on solubility for UO_4 (e.g., in vitro solubility studies in simulated lung fluids, and historical animal studies) are sparse, numerous references appear in the literature over the last 30 years regarding general solubility characteristics of industrial uranium compounds. A few specific references for UO_4 solubility studies are provided below, along with a representative list of references on solubility of industrial uranium compounds. The UO_4 and UO_3 products should be ICRP 19 Class D or W (most or moderately soluble), which is equivalent to ICRP 66 Class F or M (fast or medium dissolution). It is also of note that ICRP 54, which assigns Class W to UO_3, indicates "…there is evidence from animal studies that industrial uranium

trioxide may behave more like a class D material ". The following statements provide support for a Class D or W designation for UO_4:

- RG 8.30 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."

- Per Kathren and Burklin (2008), "…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".

- Results in Metzger et al (1997) indicated airborne U in wet process area = 97% with dissolution $T_{1/2}$ = 0.3 days and airborne U in drum load out area = 97% with dissolution $T_{1/2}$ = 0.25 days. These results might suggest the appropriateness of determining a "split DAC". However, the results of this study indicated airborne U in both the wet process and drum load out areas of 97 % dissolution with half times <0.5 day. These results are clearly indicative of a Task Group on Lung Dynamics Class D or ICRP 66 Class F compound. Several of the published studies referenced below do in fact present results suggesting "di" (2) or "tri" (3) phased dissolution patterns indicative of mixtures of uranium compounds of differing solubility classes (U_30_8 plus UO_3, e.g.). However, based on reported results, the study referenced here is clearly a single-phase dissolution pattern, i.e. single solubility class, single DAC and it is Class D. (Since the secondary "W" component is reported at only 3%, use of the < 10% exclusion rule similar to that allowable for mixtures of radionculides in air at 10 CFR 20.1204(g) would seem to be appropriate.)

Some additional references that specifically address solubility and solubility class of uranium mill and related uranium fuel cycle uranium compounds include: Scipsick et al (1985); Cook and Holt (1974); Eidson and Mewhinney (1980); Research Triangle Institute (1999); Alexander (1984); Blauer et al (1982); Brown and Blauer (1980a and b); and Moore (1984).

Potential Mixtures of Radionuclides

Studies performed in the late 1970s and early 1980s of radionuclide mobilization from several ISR operations and subsequent measurements at operating ISR operations indicate a relatively small portion of the uranium daughter products in the ore body are actually mobilized by the lixiviant (e.g., Brown, 1982 and 2007). The vast majority of secular equilibrium radionuclides remain in the host formation. This is one of the

recognized public health and safety benefits of ISR when compared with conventional milling. In these studies, Th-230 appeared to equilibrate and very little was actually removed from the host formation during ISR. The majority (80 to 90%) of the mobilized Ra-226, which was estimated to be approximately 5 to 15% 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, Pb-210 was mobilized, as the lead carbonate complexes formed in situ were virtually insoluble in the lixiviant processes studied.

In addition to the fact that very little of the uranium daughter products are mobilized during ISR, the IX resin 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 "nuclide mix" that can potentially become airborne in the precipitation, drying and packaging areas of a modern ISR is expected to be almost exclusively U-nat. In-growth of the first few short-lived daughters (Th-234 and rotactinium-234) takes more than 4 months to reach equilibrium; therefore these daughters are not expected to be associated with relatively fresh product.

It should also be noted that, in accordance with 10 CFR §20.1204(g), nuclides can be ignored in a mixture in air if the total activity is used to determine compliance with §20.1201 and §20.1502(b), so long as any nuclides ignored are < 10% of the mixture and so long as the sum of all nuclides ignored are < 30% of the mixture. For modern ISRs, such as LCI, these conditions are expected to be met.

Site Specific Information

In order to establish that natural uranium isotopes are the exclusive alpha-emitting radionuclides of concern in airborne particulate samples, LC ISR, LLC will prepare composite samples from the air particulate monitoring locations noted in **Figure 5.7-1**. These sample locations will characterize various stages in the process (e.g., lixiviant, precipitation, and drying/packaging areas). These samples will be submitted to a laboratory for radioisotopic analysis including U nat (total uranium), Th-230, and Ra-226. Results of these samples will be compared with mixture requirements in 10 CFR §20.1204(g) to ensure that the appropriate DAC from 10 CFR 20 Appendix B Table 1 is used. If necessary, a "sum of fractions rule" will be applied to establish the appropriate DAC. Time studies of job functions will be performed (or actual time workers are in process areas will be measured) and DAC–hrs of exposure estimated on weekly basis whenever air monitoring indicates workers were exposed to airborne concentrations > 10% of the DAC (which may be an "effective DAC" using sum of fractions as described above). Dose assignment will be based on the ratio of DAC-hrs of exposure to 2000 DAC–hrs /yr x 5 Rem. (see also **Section 5.7.4**, Worker Dose Calculations).

5.7.3.3 Surveys for Radon-222 and Its Decay Products

Rn-222 and its decay products can be generated and released at an in situ mine anywhere mining lixiviant or ore is open to the atmosphere. Measurements of Rn-222 decay products will be used versus direct measurements of Rn-222 because they are easier to take and because they are the best indicator of worker dose.

Measurements of Rn-222 daughters will be taken on a monthly schedule in areas where Ra-222 daughters routinely exceed ten percent of the limit or 0.03 WL above background. The RSO will consult with NRC RG 8.25 to determine if additional locations should be sampled. Rn-222 daughter measurements will be taken in at least the following locations as shown in **Figure 5.7-1**:

- near the center of the reverse osmosis bank;
- near the center of the commercial ion exchange columns;
- near the eluant make-up tank T-210A;
- near the resin handling water tank T201;
- on top of the resin shaker deck;
- near the center of the elution circuit tankage;
- near the center of the yellowcake storage tankage;
- in the laboratory; and
- in the office area.

If Rn-222 daughter concentrations are greater than 0.08 working level (25 percent of limit of 0.33 WL as per NRC Regulations found in Appendix B of 10 CFR Part 20.1001 through 20.2402), sampling frequency will increase to weekly until four consecutive weekly samples indicate the concentrations of Rn-222 daughters are below 0.08 WL. Any time the Rn-222 daughter concentration exceeds 0.08 WL, a documented inspection shall be performed by the RSO to determine and mitigate the cause of the increased concentration.

Quarterly sampling for Rn-222 daughters will also occur in areas where previous measurements have shown the daughters are not normally present in concentrations exceeding 0.03 WL (ten percent of the limit) but where proximity to sources of Rn-222 may allow them to be present. As a minimum, quarterly Rn-222 daughter sampling will occur in the following locations:

- office area,
- shop area,
- lab area, and
- raw water storage tanks.

Additionally, the RSO or HPT will collect radon daughter samples any time there is an upset condition, maintenance, or a change in operations which may result in the release of radon. For example, radon samples will be collected and analyzed before any employee will be allowed to perform a confined space entry into a vessel that has the potential to contain radon. Radon daughter samples will be collected in the immediate vicinity of the potential source of radon. When issuing an RWP, the RSO or HPT will determine the need for collecting radon daughter samples. The results of sampling will be used to ensure the employee was not exposed to elevated radon and to verify that existing procedures are ALARA.

When collecting a sample, the date, time, and status of major equipment and processes in the area will be recorded. The LLD for Rn-222 daughter measurements will be no greater than 0.03 WL (ten percent of the DAC) and shall be calculated using guidance found in Appendix B of NRC RG 8.30. Measured values less than the LLD, including negative values, will be recorded on data sheets. The LLD is set high enough to provide a high degree of confidence that 95 percent of the measured values above the LLD are accurate and do not represent false positive values. A Ludlum Model 2221 counter and Model 43-1 alpha counter with Model 180-16 sample holder or equivalent will be used to count filters.

The modified Kusnetz method will be used for measuring Rn-222 working levels. This is the standard, generally accepted method for determining radon decay product concentrations in air. One WL is that concentration of radon decay products in one liter of air that will result in the emission of 1.3×10^5 MeV of alpha energy when all of the decay products present decay to Pb-210. The Kusnetz method involves collecting an air sample for, nominally, five minutes on a high efficiency glass filter. Alpha counts on the filter will be determined by counting with an alpha scaler for one to five minutes after a decay time of 40 to 90 minutes. The total alpha disintegration rate is divided by the volume of air sampled and an empirical factor (Kusnetz factor) that relates alpha activity per liter to WL for a specified decay period. As noted in **Section 5.7.3.1**, a correction factor of 0.5 will be applied to the counting efficiency to account for any absorption by the filter.

Air samplers will be calibrated as per manufacturer recommendations or at least semiannually with a mass flow meter or other primary calibration standard. A record shall be kept of all radon surveys by the RSO until license termination and in a form compliant with NRC RG 8.7.

Environmental monitoring for Rn-222 is described in **Section 5.7.7**. The in-Plant monitoring program is summarized in **Table 5.7-1**. The specific instruments proposed to be used are listed in **Table 5.7-2**.

5.7.4 Worker Dose Calculations

Worker doses will be calculated annually based on the personal dosimeter data and the airborne radionuclide concentration measurements if the TEDE is likely to exceed ten percent of the annual dose limit as specified in Title 10 CFR, Part 20. Worker doses may be calculated at levels below the ten-percent threshold at the discretion of the RSO. NRC RG 8.30 will be used to assist with worker exposure calculations. Calculations will be based on worker exposure to both Rn-222 daughters and natural uranium. Workers who are monitored will be informed of their annual dose, and their dose records will be maintained by the licensee.

The TEDE includes the deep dose from external radiation sources plus the committed effective dose equivalent (CEDE) from intake of radionuclides by ingestion or inhalation. Doses to specific organs, committed dose equivalent (CDE), will be calculated for any worker who's CEDE exceeds 1.0 rem. Extremity doses, doses to the lens of the eye, and skin doses will be calculated for workers likely to receive an annual dose in excess of ten percent of the respective dose limits as specified in Title 10 CFR Part 20. Worker radiation doses will be calculated by methods consistent with NRC RG 8.30, 8.34, and 8.36, or in accordance with the guidance provided in the National Council on Radiation Protection and Measurements (NCRP) and/or the International Commission on Radiological Protection (ICRP) reports.

LC ISR, LLC will monitor workers by:

- use of personal dosimeters (exchanged quarterly),
- area radon decay concentration measurements (**Section 5.7.3.2**),
- area measurements of gross alpha concentrations in airborne particulate matter (**Section 5.7.3.1**), and
- measurement of radionuclide concentrations in worker breathing zones, when necessary.

Radiation doses may be calculated using the REMIT computer code or by direct calculation from monitoring data. REMIT calculates effective doses and organ doses from user-specified annual intakes and personal dosimeter monitoring data. REMIT automatically completes the NRC Form 5 that fulfills the annual reporting requirements.

Alternatively, the annual doses may be calculated directly using monitoring data and established dose conversion factors (DCFs) from Federal Guidance No. 11 (EPA, 1988), or the dose coefficients (DCs) from ICRP Report No. 68 (ICRP, 1994 and ICRP, 2001). The equations for calculating dose are given below and the DCFs and DCs for the long-lived uranium decay products are given in **Table 5.7-3**. However; since 10 CFR 20 is based on ICRP 26 and 30, the applicant must request prior approval to use ICRP 68

(NUREG 1736, Page 51 and RG 8.25, Section 4. Regulatory basis: 10 CFR 20.1204(c)). Therefore, prior to using ICRP 68, LCI will obtain prior NRC approval.

A third alternative is calculation of annual doses from inhaled radionuclides by comparison with the DAC. The DAC is the concentration of a radionuclide in air that would result in a dose of five rem to a worker exposed to that concentration for 2,000 hours. DAC values for dose calculations will be determined per the discussion in TR Section 5.7.3.2. Initially, LC ISR, LLC will assume the uranium product to be solubility class W with DAC of 3×10^{-10}. Per TR Section 5.7.3.2 , LC ISR, LLC will assess the DAC for site-specific conditions of the plant process.

In general, the in-Plant air particulate samples will be analyzed for gross alpha but assumed to be primarily, if not all, due to natural uranium. It is expected that under routine operating conditions, the only significant potential route of intake is inhalation of radionuclides in airborne particulate matter. However, as discussed in TR Section 5.7.3.2, LC ISR, LLC will verify the air particulate nuclide mix at specified plant locations in order to determine the appropriate DAC.

As noted above, the TEDE is the sum of the CEDE and the DDE derived from personal dosimetry. In the event an individual is not monitored or personal monitoring data is missing due to a lost badge, the DDE can be estimated from area exposure rate monitoring (**Section 5.7.2.2**) and worker time studies.

Routine urine bioassay measurements will be required for workers who are potentially exposed to airborne uranium. Bioassay measurements will also be required in the event of an upset condition that results in release of uranium into the Plant. Under normal conditions, the urine bioassay measurements are expected to show uranium concentrations below detection levels and are not used to estimate uranium intakes. The responses to unexpected urine bioassay measurements are given in **Section 5.7.5**. In the event the uranium concentration in urine for any worker exceeds the action level, the calculated dose based on the urine bioassay measurement will be added to the dose calculated by the methods described in this section.

The dose to the fetus of a declared pregnant worker will be calculated using the methods described in NRC RG 8.36. The dose to a declared pregnant worker must be monitored if the dose to the fetus is likely to exceed 0.05 rem. The dose to the fetus from internally deposited radionuclides must be determined if the intake is likely to exceed one percent of the Annual Limit on Intake (ALI) (If the LC ISR, LLC product is determined as Class D, then one percent is 0.02 microCuries [μCi]; otherwise ALI will be appropriately determined for Class W or mixed class product and the one percent trigger will be calculated.). Declared pregnant workers will be subject to enhanced radiation monitoring, including monthly exchange of whole body and fetal (lower torso) personal

dosimeters and calculation of the dose to the fetus from inhalation of radionuclides in airborne particulate matter, as necessary. Potential internal doses to the fetus will be determined using the methods described in NRC RG 8.36. Doses will be calculated on a monthly basis or more frequently if necessary to accurately estimate the dose to the fetus.

5.7.4.1 Dose Calculation Equations

Examples of the general equations that will be used in calculating worker radionuclide intakes and doses from routine and non-routine exposures are given in the sections below. The equations can be adapted for any exposure situation by using the specific exposure time whether it is for an entire working year or for a shorter time period. Doses calculated for special situations will be added to the routine dose received by the worker.

The symbols used in the equations are defined as follows:

C_r = concentration of radionuclide r in air (μCi/mL)
BR = breathing rate (cubic meters per hour [m^3/h])
DAC = derived air concentration (μCi/mL)
H = number of hours of exposure
DC_r = dose conversion factor for radionuclide r (Sievert per Becquerel [Sv/Bq])
I_r = annual intake of radionuclide r by inhalation (μCi)
WL = measured concentration of radon decay products in units of WL
i = exposure period
n = number of exposure periods in the year
x = number of radionuclides of interest
RnD = radon decay products
PF = respirator protection factor

Equation 1: Calculation of Long-Lived Radionuclide Intake Using Monitoring Data

(Note: Doses from inhalation of airborne particulate matter will be calculated only when the measured concentration exceeds ten percent of the DAC.)

$$I_r = \sum_{i=1}^{i=n} (BR)(C_r)(H)(10^6\ ml/m^3)\ /PF$$

The standard breathing rate used in 10 CFR 20 Appendix B, 1.2 cubic meters per hour, will be used for calculating the intake of radioactive material unless, in the professional opinion of the RSO, the breathing rate should be adjusted to reflect actual work conditions, in which case the adjusted breathing rate will be recorded with the DAC calculations. In no case will a breathing rate less than 1.2 cubic meters per hour be used in dose calculations.

In most cases, for tasks where the concentration of airborne radionuclides might exceed 10% of the DAC, worker exposures will be monitored using breathing zone samplers. Start and stop times for the sampler will be recorded. Where the worker exposure is covered by a breathing zone sample, the actual time the sampler was worn will be assumed to be the time of exposure. If a single breathing zone sampler is used to represent exposure to a group of employees, the entire group will be assumed to be exposed to the same concentrations as the individual wearing the sampler. For areas where the airborne radionuclide concentrations are likely to be greater than 10% of the DAC and the concentrations are monitored using general air samples rather than breathing zone samplers, employee time sheets will be used to estimate hours spent in specific activities where there is exposure to airborne radioactivity at concentrations likely to be greater than 10% of the DAC averaged over an 8-hour day.

Equation 2: Calculation of CEDE for Long-Lived Radionuclides by Comparison to the DAC (Inhalation Intake Only)

$$CEDE = \sum_{i=1}^{i=n} \frac{(C_r)(H)(0.0025\, rem / DAC - hr)}{(DAC)(PF)}$$

Equation 3: Calculation of CEDE for Long-Lived Radionuclides Using DCFs or DCs (Inhalation or Ingestion Intake)

Federal Guidance No. 11: CEDE = $(I_r)(DCF_r)(3.7 \times 10^4$ Bq/µCi)(100 rem/Sv)
ICRP Report No. 68: CEDE = $(I_r)(DC_r)(3.7 \times 10^4$ Bq/µCi)(100 rem/Sv)

Equation 4: Calculation of CEDE for RnD

$$CEDE = \sum_{i=1}^{i=n} \frac{(WL)(H)(5\ rem)}{(170\ h/m)(4\ WLM)(PF)}$$

Equation 5: Calculation of TEDE (Sum of CEDE and DDE)

$$\mathrm{TEDE} = \mathrm{DDE} + \sum_{r=1}^{r=x} \mathrm{CEDE}$$

Equation 6: Calculation of CDE for Critical Organs (if CEDE > 1.0 rem)

CDE = (I_r)(organ-specific DCF_r)

Note: Federal Guidance No. 11 (EPA, 1988) provides organ-specific DCFs ICRP Report No. 68 (ICRP, 1994) organ-specific DCs are included in the CD-ROM version 2.01 (ICRP, 2001)

Equation 7: Calculation of CDE for Lung Rn-222 Decay Product Exposure

Lung CDE (RnD) = (RnD CEDE)/0.12

Weighting factor for the lung = 0.12

Exposure times will be estimated using worker time studies or using recorded time spent on the task for which a RWP was issued.

The results of all exposure calculations will be maintained by the RSO until license termination and in a format compliant with NRC RG 8.7 and 10 CFR 20.2103.

5.7.4.2 Action Levels Tied to Worker Exposure Calculations

Dose calculations for workers will be performed in accordance with the guidance contained in NRC RG 8.30 and 8.34. The primary method of assigning occupational dose to workers will be via use of the stochastic inhalation ALIs and/or DACs per Methods 1 and 2, respectively, as described in RG 8.30. However, confirmed bioassay results may need to be used in cases where the estimated dose could approach or exceed ALIs and/or it is determined by the RSO that the confirmed bioassay results may provide greater accuracy or be more representative of actual intake. The methods and assumptions described in RG 8.9 will be used to estimate and assign internal dose using bioassay results

Action levels will be established for both airborne concentration surveys and for bioassay, the two methods that will be used to assess intake and therefore internal dose. At measured concentrations of 10% of the DAC, the DAC-hrs of exposure will be estimated and assigned to employees. At 25% of the DAC, the RSO will initiate an investigation to determine if corrective actions are necessary. Regarding action levels based on bioassay results, the action levels and recommended actions specified in Tables 1 and 2 of RG 8.22 will be used. (See TR Section 5.7.5 for justification for use of bioassay action levels and actions per Tables 1 and 2 of RG 8.22).

5.7.4.3 Controlling Exposure to Soluble Uranium

Intake of soluble uranium will be limited to 10 mg per week per 10 CFR 20.1201(e). Accordingly, at an assumed specific activity of 0.67 µCi/gram for U-nat (10 CFR 20, Appendix B, Footnote 3), the weekly soluble intake limit is 6.7 E-3 µCi. Initially, solubility Class W will be used to establish the appropriate ALI of 0.8 µCi and DAC of 3 E-10 µCi/ml for U natural (10 CFR 20, App B, Table 1). Assuming a 40-hour work week and average breathing rate of 20 liters/min (1.2 m^3/hr), the average concentration at the soluble weekly intake limit is approximately equal to 50% of the DAC. Compliance to this requirement will be documented by recording of worker airborne exposure in DAC –hrs, whenever long lived particulate concentrations in air are determined to be ≥ 10 % DAC and an action level of 25% DAC will be established requiring RSO investigation and potential corrective actions. Assignments of positive airborne exposure will be reviewed weekly. Accordingly, any exposures to soluble uranium > 20 % of the 10 mg/week limit will in fact be recorded (as DAC –hrs) and controlling exposure to 25% of DAC ensures both that the 10 mg / week limit is not exceeded and ALARA. Worker exposure to soluble uranium will be assessed via standard grab and breathing zone sampling particulate filtration techniques and subsequent analysis of radionuclide content of filter papers.

5.7.5 Bioassay Program

LC ISR, LLC will maintain a urinalysis bioassay program to ensure the adequacy of the airborne sampling program and to demonstrate that worker exposure to airborne uranium is ALARA and not in excess of regulatory limits. The bioassay program will be conducted in accordance with RG 8.22 and NUREG 0874. This NUREG provides the technical basis for RG 8.22. In fact, frequencies of sampling based on solubility characteristics, associated action levels and recommended actions specified in, e.g. Tables 1 and 2 of RG 8.22 are from NUREG 0874 (Section 6 compares action levels and bioassay frequencies of RG 8.22 vs. NUREG 0874). Any proposals for deviations in the Lost Creek bioassay program from the technical positions in RG 8.22 will be justified based on data derived from NUREG 0874 (or appropriate updates – see below).

Although there is some uncertainty at present regarding the applicability of TGLD solubility class D vs. Class W for Lost Creek yellowcake products, the solubility characteristics of the less soluble class W can be well within the range of dissolution half times defined by NUREG 0874 for “low temperature drying” (see NUREG 0874. Table 1-3). Additionally, data from the technical literature indicates that the UO_3, UO_4 and associated hydrates produced in modern ISRs are “low fired” and therefore relatively soluble, e.g.:

1) RG 8.30, Section 2.2: "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.

2) Kathren and Burklin (2008), Page S123: "…the more soluble compounds of uranium such as…. and UO4 are more quickly absorbed into the blood".

3) Metzger et al. (1997), Page 418: Results indicated airborne uranium 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. X ray diffraction analysis of the final dried product indicated 79% $U0_4$ * $2H_2O$, 15 % UO_3 and 3 % $CaCO_3$.

Accordingly, urinalysis as has been used for more than 30 years in uranium mills and ISRs as an appropriate bioassay method for detecting exposures to low-fired, relatively soluble uranium products. A monthly sampling frequency for workers potentially exposed to low-fired yellowcake, as recommended in RG 8.22 and NUREG 0874, is appropriate and will be used.

Although RG 8.22 requires routine bioassays for yellowcake workers at suspected inhalation exposures of ≥ 1E-10 uCi/l (one-third the Class W DAC), all workers potentially exposed to dry yellowcake will be included in the routine bioassay program and will be on a monthly urinalysis frequency. The action levels and associated recommended actions specified in RG 8.22 Table 1 and 2 will be used. Employees will submit their monthly urine samples following 1 to 2 days off from work to allow for clearance and elimination of uranium that does not become systemic and absorbed by the kidneys. Standard practice for routine urinalysis programs is to assume the 'exposure/intake' occurred on the day or days immediately following the previous sample collection. Accordingly, the action levels and actions of RG 8.22 Table 1 are appropriate based on a monthly sampling frequency. In addition to routine monthly samples, samples may also need to be collected in response to: potentially elevated airborne concentrations, as may be required by RWPs; whenever respiratory protection devices are found to be internally contaminated following use; in response to positive nasal and/or mouth swabs; or similar situations. In such cases, it will be assumed that the exposure/intake occurred at a specific time related to the activities causing the potential intake and Figure 2 of RG 8.22 will be used to establish action levels.

Footnote b of Table 1 of RG 8.22 defers to NUREG 0874 Section 6 for considerations of in vivo lung counting to detect intakes of more insoluble, high-fired materials. The NUREG recommends that in vivo capabilities should be available "to guard against the unlikely, but possible, contingency that large intakes of Class W or Y transportability might go undetected". If necessary, in vivo capabilities will be used as follow-up to

confirm urinalysis results in excess of action levels as specified in RG 8.22 Table 1y. However, as discussed above, LC ISR, LLC yellowcake products will be typical of soluble, low-fired yellowcake (UO_3/UO_4 Class D/W), and the contingency alluded to in NUREG 0874 is not applicable.

Additionally, it must be recognized that there are only a few in vivo lung counting facilities in the United States with the appropriate equipment, software and experience to measure pulmonary deposition of natural uranium at the required detection limits. These few facilities have been historically used in response to suspected "significant" intakes as based on confirmed urinalysis results. LC ISR, LLC notes that at facilities with very insoluble uranium products, e.g., high-fired oxides at fuel fabrication and/or nuclear weapon plants, fecal sampling is also used as "trigger" for in vivo analysis. However, for all but the most insoluble materials, fecal sampling is not necessary, practical nor appropriate.

LC ISR, LLC will use the basic bioassay dosimetry model and guidance in NUREG 0874 as modified by more recent elimination/retention functions (e.g., ICRP 54 and 68, and RG 8.9 and 8.34). The dosimetry model used by NUREG 0874 is the historical TGLD metabolic model (NUREG 0874, Figure 1) which assumes 67% of uranium entering blood is excreted via urine in first day without appreciable uptake to tissues; kidney uptake = 11% subsequently excreted; systemic uptake = 22% which is subsequently released to blood from which 67% excreted per day, 11% absorbed by kidney and 22% reabsorbed back to tissues. Some minor updates to this dosimetry model have been published subsequent to NUREG 0874 (e.g., ICRP 54,and 68)

Each new worker will provide a sample for an initial urinalysis. Upon termination, each worker will be required to provide an exit bioassay. Routine bioassays will be collected from each worker potentially exposed to yellowcake or directly involved in maintenance tasks in which yellowcake dust may be produced. Bioassays will also be performed if there is any reason to suspect that an inhalation exposure has resulted from exposure to an average yellowcake concentration equal to or exceeding ten percent of the DAC calculated (Section 5.7.3.2) for that radionuclide (e.g. 10% of Class W natural uranium DAC, 3×10^{-11}) or mixture of nuclides for a 40-hour workweek . If respiratory protection is used to maintain inhalation exposures below the action levels, bioassay will be performed to verify the effectiveness of the respirator. Samples will also be collected from employees who may have had an uptake of uranium resulting from an upset condition.

Samples will only be collected from workers who have not been potentially exposed to uranium for at least 36 hours and in an area free from contamination. The samples, along with three blind QA/QC samples, shall be sent for analysis at a laboratory that can detect at least five micrograms (µg) uranium per liter of urine. One blind QA/QC sample will

be spiked with ten to 20 µg/L uranium and one blind QA/QC sample will be spiked with 40 to 60 µg/L uranium. A third blind QA/QC sample will serve as a blank. If the results of the lab analysis are not within 30 percent of the actual concentration, the lab will re-analyze each of the samples provided. Additionally, the contract lab will be required to duplicate the analysis of at least ten percent of the samples, including the blanks and spikes. The lab will report the results of its QA/QC program back to LC ISR, LLC. The QC samples will comply with recommendations found in NRC RG 8.22.

Urinalysis results should be available to the RSO within 20 days after specimen collection. The contracted laboratory must report by telephone results exceeding 35 µg within 20 days after specimen collection. The RSO will be responsible for documenting compliance with the Table 1 "Corrective Actions Based on Monthly Urinary Uranium Results" found in NRC RG 8.22. The corrective actions shall be included in the Annual ALARA Report. A record shall be maintained of bioassay results and associated QA/QC until license termination and in a form compliant with NRC RG 8.7.

5.7.6 Contamination Control Program

5.7.6.1 Personnel Surveys

LC ISR, LLC will designate and post the uranium processing area of the Processing Plant area as restricted and limit access to only those individuals who have received appropriate training and/or are escorted by an experienced employee. The restricted area is shown in **Figure 5.7-1**. Signage will read, "ANY AREA WITHIN THIS FACILITY MAY CONTAIN RADIOACTIVE MATERIAL."

Routine Personnel Monitoring

Before leaving the restricted area, all individuals must perform and document an alpha survey. Individuals who have been in mine units, the byproduct storage area, or near the deep well or storage ponds will perform and document an alpha survey immediately upon entering unrestricted plant areas (e.g., offices and restrooms), before eating, or before leaving the mine site, whichever comes first. The personnel monitoring system will consist of a Ludlum Model 43 series alpha detector (Bkg. ≤ 3 cpm ; efficiency 17 -35% rated for plutonium-239) coupled to a Model 177 alarming rate meter or equivalent. The alarm will be set by the RSO after determining the efficiency of the system so that contamination above the limit will be detected. A typical alarm setting for this type of equipment is 20 counts per minute (c/m). The goal is no personal contamination above background. All workers shall receive training regarding how to properly perform and document alpha surveys. The RSO or HPT shall post by each alpha survey meter the written instructions for use of the system and the release limits in counts per minute.

All exit doors from the restricted area without a permanent or temporary scanning station will be designated and labeled as emergency exits only. A temporary scanning station may be set up for a limited period using an alpha detector/meter system approved by the RSO. Unannounced quarterly spot surveys of personnel will be performed by the RSO or HPT as recommended by NRC RG 8.30, Section 2.6. The spot surveys will take place in a non-restricted area and will include personnel who work in mine unit and process areas. The spot checks will ensure that employees perform self survey before leaving the restricted areas.

Response to Identification of Personnel Contamination in Excess of Background

Upon determination by any employee that contamination on his/her person, clothing or other personal effects exceeds background, the affected area(s) will be washed with water and soap and resurveyed. A second washing using modest abrasive methods may be required (soft brush or cloth). If the contamination remains above background, the RSO or HPT will be contacted. More aggressive methods, e.g., use of detergents, may be used but abrasion of the skin should be avoided. If the ALARA objective of background cannot be achieved without more extensive and potentially abrasive methods, the methods and release limits specified in NRC RG 8.30, Section 2.6 would be used and all detected activity would be assumed to be removable. If these limits cannot be achieved w/o abrasion of the skin or other potentially harmful impact to the employee, the RSO may need to refer the employee for medical intervention.

Since any beta – gamma contamination at an ISR (or U mill) must be associated with alpha emitting nuclides, no special monitoring or survey for beta – gamma emitters are required. The lack of detectable alpha contamination assures no beta – gamma contamination. For example, the short-lived beta-gamma progeny protactinium and Th-234 take approximately four months to reach equilibrium and little would be associated with fresh yellowcake product and they must be associated with their uranium parent. Another potential contaminant, Ra-226, emits alpha particles at more than 94% yield. Additionally, since the personnel "release objective" is background (or very close to it, e.g., RG 8.30 Section 2.6 limits), the quantification of potential contribution from other uranium series nuclides is neither dosimetrically significant nor relevant.

However, surveys performed with a pancake GM probe (e.g., Ludlum Model 44 – 9 (as described in **Section 5.7.6.2** for area surveys) will detect any significant beta contamination present. Since release limits for beta / gamma emitters are identical to alpha (" uranium and daughter products" @ 1000/5000 dpm per 100 cm2), use of these limits for total activity are protective (e.g., NRC RG 1.86 and FC 83-23 as referenced in RG 8.30 – see discussion in **Section 5.7.6.3**).

Although the objective of personnel decontamination is background in accordance with NRC RG 8.10 and 8.31, should background not be achievable without potential damage to the skin of the affected employee, the approach and limits described in NRC RG 8.30, Section 2.6 will be applied. No requirement is specified therein to establish the nuclide mix of the ≤ 1000 dpm / 100 cm^2 limit (5000 dpm/ 100 cm^2 for soles of shoes) nor is their any technical basis for doing so since there is no need at levels approaching background to perform a "dose assessment".

5.7.6.2 Area Surveys

Surface contamination will be assessed in Plant areas and regularly evaluated by visual inspection and measurement. Surface contamination will be kept to a minimum in the locations where work with uranium is performed and uranium compounds may accumulate, such as the ion exchange, resin slurry, and filter press areas. NRC regulations do not specify a limit on surface contamination in restricted areas. However, surface contamination will be managed and minimized in these more routinely contaminated portions of the restricted area to reduce the potential for resuspension of uranium dust that can result in inhalation or ingestion intake.

Unrestricted areas of the Plant (e.g. offices, laboratory, etc.) and restricted areas where work with uranium is not performed will be surveyed (spot checked) weekly for removable contamination (smear surveys). The goal for these areas is background. Areas that exceed the removable contamination limit of 1,000 dpm alpha or beta per 100 cm^2 will be cleaned immediately and re-surveyed. Interim instrument surveys for total contamination may be performed. If the total contamination level exceeds the removable contamination limit, the removable contamination level will be determined using smears. In any case, areas demonstrating removable contamination in excess of the contamination limits will be cleaned and resurveyed

Objectives regarding acceptable levels of contamination in all restricted and controlled areas (area within security fence) of the facility are ALARA. Good housekeeping practices, including regular wash downs of potentially contaminated process areas, will reduce potential for airborne exposure and contamination of persons. Routine and ad-hoc air sampling in process areas and contamination assessment / decontamination practices at restricted area egress points will maintain exposures of personal from potential intakes ALARA. Objectives for contamination in all areas of the LC plant, ancillary facilities and anywhere else within the controlled (security fenced area) and restricted areas will be ALARA applying principles articulated in NRC RG 8.10 and 8.31.

For instrument surveys for total contamination, LCI will use of a Ludlum Model 2224 counter and model 44-9 GM pancake probe (or equivalent) with an alpha efficiency of 15% (rated for plutonium-239) and background of 60 cpm and/or a Ludlum 43 series

alpha detector (or equivalent) which has reported alpha efficiencies (rated for plutonium-239) of 17 to 35 % with alpha background ≤ 3 cpm. Both types of instruments will be on hand, and usage will depend upon area and product contaminant mix determined from weekly smear surveys. Surveys performed with a pancake GM probe (e.g., Ludlum Model 44 – 9 (as described in section 5.7.6.2 for area surveys) will detect any significant beta contamination present. Since release limits for beta / gamma emitters are identical to alpha (" uranium and daughter products" @ 1000/5000 dpm per 100 cm2), use of these limits for total activity are protective (e.g., NRC RG 1.86 and FC 83 -23 as referenced in NRC RG 8.30 – see discussion in section 5.7.6.3).

5.7.6.3 Material Release Surveys and Limits

Methods and Procedures

The RSO or HPT will survey potentially contaminated items before they are released from the facility. Items, which cannot be representatively surveyed due to geometry or any other reason, may not be released for unrestricted use. A Ludlum Model 2224 counter and Model 44-9 pancake GM probe or equivalent as described above will be used for release surveys. Survey equipment shall be calibrated as per manufacturer specifications and at least annually. Instruments used to assess surface contamination shall be checked for proper response daily when the plant is operating. Alpha survey instruments used for personnel surveys shall be response checked before each survey to ensure they are in working order. Additional operational checks on alpha meters shall be performed weekly.

Equipment and surfaces shall not be painted over for the purpose of meeting release criteria. However, if painting over an area with contamination that cannot reasonably be removed is determined by the RSO to be ALARA, it may be allowed as long as the contamination on the article or surface is characterized and documented. The item must be visibly labeled as contaminated. Adequate records will be maintained to ensure that the article or surface is not inadvertently released for unrestricted use.

Equipment that cannot be adequately surveyed for contamination will not be released for unrestricted use.

LC ISR, LLC will ship yellowcake slurry to other facilities for drying and packaging. Prior to the release of packages containing yellowcake, the packages shall be washed and thoroughly surveyed to ensure compliance with DOT release standards found in 49 CFR 173.433(a) and (b).

Contamination Limits to be Applied for Release of Equipment and Materials From Restricted Areas

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. Studies performed in the late 1970s and early 1980s of radionuclide mobilization from several ISRs (Brown, 2007 and Brown, 2008) 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. The majority of the mobilized Ra-226 (80 to 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 (e.g., as 11(e)(2) byproduct material). It is only in this material in which Ra-226 concentrations would be expected to be elevated relative to equilibrium with uranium but not w/o some uranium and potentially small amounts of its other short-lived daughter products.

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 as described in NRC RG 8.30, Section B (*Discussion)* indicates, "The contents of this guide conform with NRC's current licensing practice". LC ISR, LLC is not aware 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.

Recommended surface contamination limits are defined in Table 2, RG 8.30. A footnote to RG 8.30 Table 2 indicates the stated contamination levels are taken from RG 1.86 and from USNRC (Aug 1987). LC ISR, LLC also notes that NRC's Policy and Guidance Directive FC 83-23, uses the 1982 version of USNRC (Aug 1987) as its ENCLOSURE 2 with the identical radionuclide categories and contamination limits as the 1987 version as well as with RG 1.86. The 1987 document is essentially identical to the 1982 version referenced in FC 83-23. Accordingly, FC 83-23 and both the 1982 and 1987 versions of USNRC, 1987, use identical radionuclide categories and quantitative limits although the 1987 document also specifies dose rate guidance (mrad/hr for beta gamma emitters). Therefore the radionuclide categories, limits and intended application of FC 83-23, NRC (1982 and 1987), RG 8.30 and RG 1.86 are all consistent.

Since the title of NRC 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 is was clearly intended to be applied to uranium recovery facilities with expected varying degrees of equilibrium and ratios of natural uranium series radionuclides. Nothing in the historical

documents referenced above provide any contradiction to or clarification of this interpretation. The use of the phrase " and associated decay products" in e.g., FC 83-23 Enclosure 2 (and subsequent 1987 revision) does not appear to be defined nor clarified in any historical documents nor is there any indication of distinction relative to the phrase " and its daughters" as used in NRC RG 8.30.

It is LC ISR, LLC's understanding that NRC's SECY 98-155 contains examples of dose calculations resulting from the FC 83-23 / RG 1.86 contamination limits. 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. LC ISR, LLC assumes that the data provided in this attachment is similar or identical to the dose calculations NRC presented in NUREG/BR0241 (1997, also no longer available*)*. In that document the following results were presented for each of the groupings at the average surface activity guidelines of RG 1.86 (as reproduced in Abelquist, 2001)):

U-nat, U-235, and U-238 and daughters	13 mrem/yr
Ra-226, Ra-228, Transuranics	0.2 mrem/yr
Th-nat, Th-232, Sr-90	28 mrem/yr
Beta-gamma emitters	20 mrem /yr

It is interesting to note that three of the groupings are generally consistent with NRC's 25 mrem/yr criteria in 10 CFR 20 Subpart E and NUREG 1757 as used for the risk/dose based approach under the License Termination Rule. The estimated dose of 0.2 mrem /yr for the radium / transuranic grouping indicates that the RG 1.86 guideline of 100 dpm / 100 cm^2 for those radionuclides should be increased by approximately a factor of 100 to yield the same dose. Admittedly, the exposure scenarios and modeling assumptions NRC used are unknown.

Although Ra-226 concentrations can be elevated relative to equilibrium with uranium in certain process precipitates, 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 in "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 by NRC indicate they are protective and provide an appropriate standard of care. Accordingly, the applicable recommendations provided in RG 8.30 incorporating the guidance contained in "Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct Source or Special Nuclear Material" (1987) will be integrated into the contamination assessment and control elements of the Lost Creek ISR radiation protection program. Upon official published

revision of the current NRC regulatory guides and standards of practice for release criteria to unrestricted areas applicable to source material facilities, LC ISR, LLC will revise the project procedures accordingly.

5.7.6.4 Inspections

A daily inspection shall be performed in the Plant, when it is operating, by the RSO, HPT, or designated, trained worker to check for proper containment of yellowcake and mining solutions, proper storage of PPE, radiation protection signage, access control, and security measures. All visible contamination shall be cleaned up immediately. At least once per week, the plant supervisor shall accompany the RSO on an inspection.

Weekly inspections of rooms where work with uranium is not performed shall be conducted and documented by the RSO or HPT. During the inspection, direct alpha readings will be taken (via survey instrumentation as described in **Section 5.7.6.2** above) in at least the control room, lunchroom, and change rooms. If the surface contamination in these areas exceeds those listed in Table 2 of NRC Regulatory Guide 8.30, the RSO or HPT shall conduct and document an investigation. If survey results indicate total alpha contamination greater than 1000 dpm/100 cm^2, a smear survey will be performed to determine the level of removable alpha contamination. Any area with greater than 1000 dpm/100 cm^2 of removable alpha contamination will be cleaned and resurveyed. Any object contaminated with greater than 5,000 dpm/100cm^2 total alpha will be decontaminated before being allowed back into a non-restricted area. Mine unit inspections are described in detail in **Section 5.7.6.6**.

5.7.6.5 Standard Operating Procedures

SOPs will be developed for major components and equipment and for processes which may present a hazard to the worker, the environment, or the operation if performed incorrectly. SOPs for processes involving radionuclides must be approved by the RSO and will include a section discussing appropriate PPE, such as gloves, coveralls, boots, etc. SOPs will discuss possible upset conditions and how the employee should respond. Each SOP will be tracked to ensure the appropriate version is in distribution. For work in radiological areas not covered by an existing, approved SOP, a Radiation Work Permit (RWP) will be prepared by the RSO (or designee) and approved by the Plant Supervisor, which will define any necessary PPE, radiological controls and measurements necessary to ensure that the work can be accomplished maintaining exposures below limits and ALARA.

5.7.6.6 Plant and Mine Unit Control

In order to prevent spills of mining solutions, the following precautions will be taken.

- Piping and associated fittings will only be constructed of materials that are chemically compatible, able to withstand the expected operating pressures, and compatible with ambient conditions.
- Mine unit pipelines and manifolds will be pressure checked before being placed into operation and after significant repairs.
- Each operating mine unit will be inspected at least once per day in order to identify and correct operational problems. The entire Plant also will be inspected at least daily when operating (see section 5.7.6.4 above).
- Automated monitoring will be installed in each header house so any significant change in flow or pressure is detected and operators are notified. When automated systems are inoperable, each header house will be inspected at least twice per day.
- Notification of excursions and investigation reports of excursions.

Each operating header house will be inspected at least once per day by the operations staff with the results documented. The inspector will look for the following:

- Leaks of lixiviant in the header house;
- Failing pipes and fittings;
- Conditions that may lead to a release of lixiviant;
- Proper capping of wellheads and pipes that are not in use;
- Exposed scale that could become airborne;
- Exposed piping that is supposed to be buried;

Any condition discovered during the inspection that may lead to the spread of contamination will be repaired in a timely manner or made safe. Results of the inspection will be made available to the RSO and will be maintained for the life of the license.

Impacts and Response to Spills in the Mine Units

It is reasonable to assume that the "worst case" spill scenario involving mine unit and/or satellite plant operation would involve spill of pregnant lixiviant or loaded resin. In its risk analysis in NUREG 6733, NRC assumes the pregnant lixiviant and loaded resin contains Ra-226 at a concentration of 3.4E3 pCi/L and U-nat at a concentration of 1.7E5 pCi/L. The short-lived decay products of Rn-222 were assumed to be in equilibrium with the Ra-226. It was assumed the impacted area would be cleaned up immediately upon recognition (which LC ISR, LLC will do).

LC ISR, LLC calculated the criteria for cleanup assuming a maximum annual dose to a worker of 100 mrem per year and 25 mrem per year for a member of the public. A RESRAD analysis (RESRAD Version 6.3. Argonne National Laboratory) 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 (**Attachment 5.7-4**). 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/m^3, (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. LC ISR, LLC 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, 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 and 2.2 pCi/g Ra-226 above background. LCI proposes this as the initial ALARA target decommissioning criterion for a spill of this nature. However, LC ISR, LLC will use RESRAD with appropriate current land use and process specific estimates of spill concentrations of Ra-226 and U-nat to re-determine the appropriate and justifiable cleanup criterion for decommissioning. Regardless, the criteria will meet the decommissioning "Radium Benchmark" of 10 CFR 40, Appendix A, Criterion 6.

In the event of a spill of mining solutions in the mine unit, LC ISR, LLC will recover as much of the fluid as possible using equipment designated for this purpose. The RSO or HPT will be notified of the spill so they can visit the site and perform an assessment of the radiologic risks. The assessment will include:

- Drawing of the affected area so the location can be identified at decommissioning;
- Determination of the amount of fluid spilled;
- Calculation or analysis to determine the concentration of radionuclides in the soil;
- Determination of the cause of the spill;
- Determination of safety precautions that need to be taken immediately, if any;
- Determination of the extent and timing of soil cleanup; and

- Determination of whether or not reporting is required pursuant to 10 CFR 20.2202 (immediate notification within 24 hrs) and 20.2203 (written report within 30 days) and 10 CFR 40.60 (4 hrs 24 hr immediate notification vs. written report within 30 days). If reporting is necessary, the RSO shall complete the report in the time frame designated in the applicable regulations.

The RSO or HPT may call upon the expertise of the area supervisor to assist with the assessment. The results of the assessment will be filed in a decommissioning file until the license is terminated.

At least once per year, the Manager of EHS and Regulatory Affairs will convene the ALARA Committee to review the cause of recent spills. The ALARA Committee will consist of at least three individuals with experience in operations. After reviewing the causes of recent spills, the Committee will send a report to mine management detailing reasonable recommendations on how to prevent and minimize the size of future spills.

5.7.6.7 Reports and Records

The following records related to contamination control will be maintained for the life of the license:

- Daily RSO inspections of the Plant;
- Weekly RSO inspections of non-process areas;
- Spot checks of personnel surveys;
- Material release surveys, including transport of yellowcake slurry;
- Calibration and function checks of survey instruments used for material release, personnel surveys, and surface contamination surveys;
- Spill assessments;
- ALARA Committee spill findings;
- Daily well field inspections; and
- ALARA audits.

These reports will be completed as required, following the appropriate regulations and company policy, and maintained for the life of the license:

- Reports related to loss of control of material pursuant to 10 CFR 20.2202 and 20.2203 and 10 CFR 40.60;
- Semi-annual Effluent Report pursuant to 10 CFR 40.65;
- Internal Monthly RSO Report;
- Internal Spill Assessments; and
- Internal Annual ALARA Report;

- Land Use Survey;
- Environmental Monitoring Data;
- Corrective Action Reports; and
- SERP Information.

5.7.7 Airborne Effluent and Environmental Monitoring Programs

As noted in Title 10 CFR Part 20, Subpart L, Section 20.2103, "Records of Surveys", the Airborne Effluent and Environmental Monitoring Program will:

- maintain records showing the results of surveys and calibrations required by Title 10 CFR Part 20, Subpart L, Sections 20.1501 and 20.1906, and
- retain these records until license termination.

These retained records will include:

- the results of surveys to determine the dose from external sources and used in the assessment of individual dose equivalents;
- the results of measurements and calculations used to determine individual intakes of radioactive material and used in the assessment of internal dose;
- the records showing the results of air sampling, surveys, and bioassays required pursuant to Title 10 CFR Part 20, Subpart L, Sections 20.1703(c)(1) and (2); and
- the results of measurements and calculations used to evaluate the release of radioactive effluents to the environment.

5.7.7.1 Operational Radiation Monitoring

Airborne Effluent

Airborne effluent and environmental monitoring programs will be carried out as recommended in the NRC RG 4.14. Air particulate samples will be collected continuously at the five Air Particulate Sampling Locations and used to determine U-nat, Th-230, Ra-226, and Pb-210. Air filters will be collected weekly and analyzed quarterly. As a result, should yellowcake drying and packaging operations be initiated at the plant, air particulate monitoring stations are already established.

In agreement with RG 4.14, during operations LC ISR, LLC will monitor radon gas and passive gamma radiation at the same locations as in the 2010-11 radon and passive gamma baseline monitoring (**Section 2.9.4** and **Figure 2.9-27**). Of the 2010-11 radon

and passive gamma sampling sites, five (PR-1, -2, -3, -5, and -10) will be co-located with radiological air particulate samplers; four (PR-7, -8, -9, and -11) will be located at the 2006-08 radon and gamma sites not associated with air particulate samplers; and three additional sites (PR-4, -6, -12) will be sampled based on 2007-08 and 2009 MILDOS analyses or on-site wind pattern (refer to **Section 2.9.4** for a discussion of these locations). Thus, there will be a total of twelve locations, **Figure 2.9-27**, with radon gas and direct passive gamma radiation measurement

Radon gas will be monitored continuously with quarterly analysis using alpha track etch detectors or equivalent. The device shall be able to accurately detect down to 0.33 pCi/L based on a 90-day sample. At least one location shall have two monitoring devices as part of QA/QC.

Passive gamma will be measured continuously with quarterly analysis using passive integrating devices or an equivalent. The device shall have a range of at least one mrem to 500 rem with an accuracy of at least plus or minus 15 percent. At least one location shall have two monitoring devices as part of QA/QC.

Operational sampling will be periodically assessed to determine if additional sampling locations will improve LC ISR, LLC's ability to comply with regulatory intents. Background conditions will be considered to be primarily represented by monitoring at HV-3 and PR-7 (Figure 2.9-27), but also HV-5 (upwind but closer proximity to plant site).

Vegetation

With regard to RG 4.14, 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. Baseline vegetation sampling, as described in **Section 2.9.3.2** has been conducted to facilitate final site restoration.

Soils

There will be no significant release of airborne particulate radionuclides during production since the LC ISR, LLC 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. Po-214 will decay rapidly to Pb-210. The 2009 'near Plant' MILDOS analysis (**Section 2.9.3.1**) indicates that the maximum approximate LC ISR, LLC Permit Boundary surface contamination of 5,000 pCi/m^2 is 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/cm^3. This incremental concentration would be indistinguishable from background. Background Pb-210 concentrations range from non-detect at a detection limit of 0.1 pCi/g to 4.9 pCi/g (**Table 2.9-1**).

Nevertheless, soil sampling will be conducted annually during operations at the five Air Particulate Sampling Locations and analyzed for U-nat, Th-230, Ra-226 and Pb-210 per RG 4.14. Radionuclide particulates including the long lived radon progeny Pb-210 that are detected will be compared to baseline values to assess impacts and/or undesirable trends. As described in **Section 2.9.1.1**, during the preoperational radiological baseline characterization program, surface soil sample analyses were conducted per RG 4.14 recommendations. Those recommendations indicate that, in addition to Ra-226 analysis for all soil samples, ten percent of samples should be analyzed for U-nat, Th-230, and Pb-210. All ten of the correlation grid samples were analyzed for these additional radionuclides, providing a reasonably representative characterization across the Permit Area, including baseline concentrations of Pb-210. Per RG 4.14, soil sampling will be conducted annually at each of the five air particulate monitoring locations, which have already been placed in service, for monitoring of Pb-210 deposition and in anticipation of a yellowcake dryer amendment request.

Public Doses

Calculations performed in accordance with existing NRC guidance will be used to estimate radionuclide source terms (including radon) and calculate off-site dose to the public. For example, Section 2.6 in Regulatory Guide 3.59 provides methods acceptable to NRC for estimating the radon source term during ISR operations. Additionally, NUREG 1569, Appendix D, provides the MILDOS AREA methodology acceptable to the NRC, which includes expressions for calculating the annual Rn-222 source terms from various aspects of ISR operations.

LC ISR, LLC believes the requirements for providing a semi-annual report to NRC per 10 CFR 40.65 of the quantity of each of the principal radionuclides released to unrestricted areas can be met through these methods since 10 CFR 40.65 does not specifically require "measurement". Furthermore, the disperse and diffuse nature of potential radon releases from multiple locations at ISRs makes empirical measurement impractical. Throughout the 30 years of ISR operational experience in the US there is no evidence of public exposure from radon releases in excess of public exposure criteria. For

example, NUREG 1910, Table 4.2-2 presents nine dose estimates to offsite receptors solely from radon releases from ISR facilities, all of which are ≤ 40 mrem/yr. Further, NUREG 1910, Section 4.2.11.2.1 states " all doses reported are well within the 10 CFR 20 annual radiation dose limit for the public of 1 mSv (100 mrem / yr)".

The results of the Airborne Effluent and Environmental Monitoring Program shall be reported to the NRC semiannually as required by 10 CFR 40.65.

5.7.8 Groundwater and Surface Water Monitoring Programs

Sampling to evaluate the radiological impacts of ISR is part of the overall operational program (**Section 3.0**). Baseline conditions have been assessed on both a regional and site-specific basis to provide information on the overall quantity and quality of water in the Permit Area and its vicinity (**Section 2.7**). Operational monitoring has been designed to identify any potential impacts to hydrology of the Permit Area and its vicinity during both ISR activities and groundwater restoration. This operational monitoring includes sampling of private wells and monitor wells specific to individual mine units. This sampling information augments the information on production and injection control, such as injection rates and pattern balance, which is instrumental to efficient ISR (**Section 3.2.7**).

During baseline and stabilization water well sampling, all wells will be purged of at least three casing volumes before the final sample is collected. A casing volume is equal to the volume of casing below the water table. The water to be sampled must also be stable with regard to the field parameters of temperature, pH, and conductivity. A total of three stability samples will be collected no closer than ten minutes apart. The water will be considered stable if there is less than a 10% change between each of the three samples for each of the analytes.

During operational sampling of mine unit monitor wells, the well will be purged at least one casing volume before collecting a sample. The purge volume is relaxed during operations because the wells are sampled frequently which prevents the water in the casing from becoming stale. The stability sampling will be the same as above to ensure that the formation water is being collected. If private wells are sampled, the sampling protocols described above will be followed

Samples will only be collected in clean, non-reactive containers. The samples will be kept cool (not frozen) until analyzed and preserved using WDEQ guidelines if the analysis will not be performed within two days. The Quality Assurance Program shall describe a rigorous system of checks to ensure samples are being collected and analyzed properly.

Before purging any well, the water level will be measured (if the wellhead design allows access) to within 0.1 feet and the data will be recorded and maintained for the life of the facility.

5.7.8.1 Baseline Conditions

Groundwater and surface water monitoring have been conducted to establish baseline conditions, including the types of radionuclides present, if any, and their concentrations. The results and analysis of the baseline monitoring are presented in detail in **Section 2.7** and are summarized here.

Quarterly water level measurements and water quality samples have been collected in 25 monitor wells and one water supply well (completed in the DE, LFG, HJ and UKM Horizons). The sampling dates for each round are listed in Table 2.7-13. Wells MB-7 and MB-10 were completed in the DE Sand but there is insufficient water for sampling.

- DE Monitor Wells: LC29M, LC30M, LC31M, and MB-1;
- LFG Monitor Wells: LC15M, LC18M, LC21M, LC25M, MB-2, MB-5, and MB-8;
- HJ Monitor Wells: LC16M, LC19M, LC22M, LC26M, MB-3B, MB-6, and MB-9; and
- UKM Monitor Wells: LC17M, LC20M, LC23M, LC24M, LC27M, LC28M, and MB-4

The water samples were submitted to a contracted laboratory; and the analytical results were evaluated for QA and then input into the Lost Creek digital analytical water quality database for evaluation The sample results from these 25 wells indicate that the groundwater within the shallow Battle Spring aquifers beneath the Permit Area is a calcium-sulfate to calcium-bicarbonate type. There appears to be some variability in water chemistry; but, overall, there is no significant difference in the major water chemistry between the production zone and the overlying and underlying aquifers.

In general, groundwater in the shallow Battle Spring aquifers within the Permit Area tends to have a relatively good water quality, with the exception of the presence of radionuclides. Ra-226 plus Ra-228 exceed the EPA MCL in over 80% of the samples collected; and the average uranium concentration is an order of magnitude greater than the EPA MCL for that constituent. These elevated radionuclide concentrations are consistent with the presence of naturally occurring uranium ore within the aquifer.

5.7.8.2 Operational Monitoring

Surface Water Monitoring

The Lost Creek Permit Area has three shallow ephemeral drainages within the permit area. The only time these drainages contain water is immediately after a heavy rain storm or during snow melt. The Lost Creek Project is located in high desert terrain which receives minimal precipitation. For example, the weather station at nearby the Lost Soldier property recorded less than five inches of precipitation in the 2006 to 2007 monitoring period (Figure 2.5-2a). LC ISR, LLC does not propose to perform routine surface water sampling simply because surface water is rarely present. However, if a spill impacts a drainage, an automatic sampler will be installed in the downstream and upstream channel to quantify the radionuclide content of the water during the next precipitation event that results in flow in the channel. The upstream sampler will serve as a background measurement.

The four existing impoundments within the permit area and immediate vicinity, which are described in Section 2.7.1.1 do not contain water except on rare occasion, so no impoundment sampling is proposed during operations. Three of these ponds are not subject to drainage from the Permit Area. The fourth stock pond is on an ephemeral drainage that drains the easternmost portion of the Permit Area, with limited operations upstream. Similar to the drainage sampling described above, if there is a spill that impacts the impoundment, a sample will be collected during the next precipitation event that fills the pond.

Private Well Monitoring

There are no drinking water wells or agricultural water wells within the Permit Area or within 1.24 miles (two kilometers) of the Permit Area. There are also no stock wells within the Permit Area. The operational BLM stock wells within 0.62 miles (one kilometer) of the Permit Area (**Section 2.2**) will be sampled on a quarterly basis with BLM's consent. Groundwater samples will be collected in accordance with the instructions contained in LC ISR, LLC's Environmental Manual. Samples will be analyzed for U-nat and Ra-226, and records of the sampling results will be maintained until license termination.

Life-of-Mine Wells

The 25 wells listed in **Section 5.7.8.1,** which were used to establish baseline conditions, will be available for the life-of-mine. Water level measurements will be taken in these wells quarterly, and other samples may be collected from these wells depending on the development of mine units near or encompassing the wells.

Disposal Well Monitoring

The disposal well(s) will be operated under the conditions specified in a WDEQ Class I Permit. At a minimum, hourly monitoring will include: annulus pressure, injection pressure, flow rate leaving the processing Plant, and flow rate at the injection well. An employee will walk the length of the buried pipeline leading from the Plant to the wellhead at least once per week to look for leaks. The wellhead will be inspected at least once per day by a trained employee, with the results documented.

Monitoring Specific to Mine Units

During ISR, the monitoring program for each mine unit is designed to detect excursions of lixiviant from the pattern area, and it includes the monitor ring wells completed in the same sand as the pattern area (the HJ Sand) and monitor wells in overlying or underlying water-bearing strata. The overlying monitor wells will be completed just above the aquitard above the HJ Sand. The underlying monitor wells will be completed just below the aquitard underlying the HJ Sand. Neither the overlying nor underlying monitor wells will have any part of their completion within the HJ Sand. Excursion detection is based on comparison of concentrations of specific parameters with the Upper Control Limits (UCLs) for those parameters, which are calculated from the baseline concentrations of those parameters. During restoration, the monitor program is designed to ensure restoration activities are proceeding as planned. Restoration success is based on comparison of concentrations of specific parameters with class-of-use standards and baseline concentrations for those parameters.

Mine Unit Baseline Water Quality and Upper Control Limits

After delineation of a pattern area, monitor wells will be installed around that area as described in **Section 3.2.2**. A pump test will be used to verify communication between monitor wells in the monitor ring and the pattern area and lack of communication between the pattern area and overlying and underlying monitor wells. Baseline groundwater samples will be collected in accordance with the instructions contained in LC ISR, LLC's Environmental Manual.

All the mine unit monitor wells will be sampled at least four times at least 14 days apart. One round of samples will be analyzed for the parameters listed in **Table 5.7-4** and three rounds will be analyzed for just the UCL parameters. UCLs will be set for parameters that would be indicative of a migration of lixiviant from the mine unit. It is anticipated that these parameters will be chloride, conductivity, and total alkalinity. Chloride is a common UCL in Wyoming due to its low natural levels in the native groundwater and because chloride is introduced into the lixiviant from the ion exchange process (uranium

is exchanged for chloride on the ion exchange resin). Chloride is also a very mobile constituent in the groundwater and will show up quickly in the case of a lixiviant migration to a monitor well. Conductivity is another common UCL because it is an excellent general indicator of overall groundwater quality. Total alkalinity concentrations should be affected during a potential excursion, as bicarbonate is the major constituent added to the lixiviant during mining. UCLs will be set at five standard deviations to the baseline average for the indicator.

Excursion Detection

Excursion detection will consist of sampling the monitor wells at least twice per month, and no less than ten days apart, and analyzing the samples for the UCL parameters. The monitor wells will be sampled as per the above schedule, except in the event of certain situations. These situations include inclement weather, mechanical failure, holiday scheduling, or other factors that may result in placing an employee at risk or potentially damaging the surrounding environment. In these situations, LC ISR, LLC will document the cause and the duration of any delays. In no event shall a delay be greater than five days. Records of UCL monitoring, including chemical assays, shall be maintained until license termination. Although not an excursion indicator, water levels will be obtained and recorded prior to each well sampling

Excursion Verification and Corrective Action

During routine sampling, if two of the three UCL values are exceeded in a monitor well, or if one UCL value is exceeded by 20 percent, the well will be re-sampled within 24 hours and analyzed for the excursion indicators. If the second sample does not exceed the UCLs, a third sample will be taken within 48 hours. If neither the second or third sample results exceed the UCLs, the first sample will be considered in error.

If the second or third sample verifies an exceedance, the well in question will be placed on excursion status. Upon verification of the excursion, the NRC Project Manager will be notified by telephone or email within 48 hours and notified in writing within 30 days.

If an excursion is verified <u>during operations or active groundwater restoration</u>, the following methods of corrective action will be instituted (not necessarily in the order given), dependent upon the circumstances.

- A preliminary investigation will be completed to determine the probable cause.
- Production and/or injection rates in the vicinity of the monitor well will be adjusted as necessary to generate an effective net process bleed, thus forming a hydraulic gradient toward the production zone.
- Individual wells will be pumped to enhance recovery of ISR solutions.

- Injection into the pattern area adjacent to the monitor well may be suspended. Recovery operations will continue, thus increasing the overall bleed rate and the recovery of the ISR solutions.

In addition to the above corrective actions, the sampling frequency of the monitor well on excursion status will be increased to weekly. An excursion will be considered resolved when the concentrations of excursion indicators do not exceed the criteria defining an excursion for three consecutive one-week samples and NRC has been notified by mail. If an excursion is not corrected within 30 days, a sample will be collected and analyzed for parameters listed in WDEQ-LQD Guideline 8 Appendix I Sections IV and VA(1) and the applicable EPA MCLs. Once parameters no longer exceed the UCLs, a final sampling and analysis of the WDEQ-LQD Guideline 8 parameters will be performed.

The following methods of corrective action for an excursion occurring during the restoration stability monitoring period will be instituted (not necessarily in the order given), dependent upon circumstances.

- A preliminary investigation will be completed to determine the probable cause and the area affected.
- Affected wells will be analyzed for the parameters listed in WDEQ-LQD Guideline 8 Appendix I Sections IV and VA(1) and the applicable EPA MCLs.
- An assessment will be performed to determine what actions, if any, need to be taken to protect the groundwater outside the exempted aquifer (USDW). If sufficient data to make such a determination is not available, additional wells may be installed to fill in data gaps.
- If the excursion may result in a degradation of a USDW, a pump back or pump and treat plan will be initiated as soon as possible to recover the excursion. The stability monitoring period will continue but will not be considered successful until Section 6.2.4 of the TR is completed and the excursion is recovered or it can be demonstrated that the remnant of the excursion will not degrade the water quality of the surrounding USDW.
- If the excursion will not result in the degradation of a USDW, then the stability monitoring period may continue. However, at the end of the successful stability monitoring period the wells affected by an excursion will be analyzed for the parameters listed in WDEQ-LQD Guideline 8 Appendix I Sections IV and VA(1) and the applicable EPA MCLs to verify the previous assessment that the USDW will not be degraded.

LC ISR, LLC will terminate lixiviant injection in the area of an excursion or provide additional reclamation surety, that is agreeable to the NRC, if an excursion cannot be remediated within 60 days of confirming the excursion.

Restoration Criteria

During restoration activities, monitoring for excursion detection will continue as outlined above to ensure that none of the restoration activities, such as reinjection of RO permeate, result in unanticipated migration of fluids from the pattern area.

Once restoration activities are near completion, sampling to verify restoration success will be initiated. The applicable restoration criteria for each mine unit and the sampling necessary to verify restoration success are described in **Section 6.2.**

5.7.8.3 Storage Pond Leak Detection Monitoring

The Storage Ponds will be lined and equipped with a leak detection system. During operations, the leak detection standpipes will be checked for evidence of leakage. Visual inspection of the pond embankments, fences and liners and the measurement of pond freeboard will also be performed during normal operations. A Pond Inspection Program will be developed for the Project and will meet the guidance contained in NRC RG 3.11 and commitments made in **Section 5.3.2** of this application.

A minimum freeboard of three feet will be maintained for any Storage Pond during normal operations. Anytime six inches or more of fluid are detected in a leak detection system standpipe, it will be analyzed for specific conductivity. Should the analyses indicate that the liner is leaking (by comparison to chemical analyses of pond water), the following actions will be taken.

- NRC will be notified by telephone or email within 48 hours of leak verification.
- The level of the leaking pond will be lowered by transferring its contents into an adjacent pond. While lowering the water level in the pond, inspections of the liner will be made to determine the cause and location of the leakage. The area of investigation first centers on the pond area specific for the particular standpipe that contains fluid.
- Once the source of the leakage is found, the liner will be repaired and water may be reintroduced to the pond.
- A written report will be submitted to NRC within 30 days of the leak verification. The report will include analytical data and describe the cause of the leakage, corrective actions taken, and the results of those actions.

5.7.9 QA Program for Radiological Monitoring Programs

The operations plan follows a prescribed radiological monitoring program that will generate reasonably valid data of a defined quality. The QA Program will be designed to:

- identify sampling and measurement processes with deficiencies and report them, and
- obtain confidence in the monitoring program results to determine valid data.

Steps of the monitoring process will involve QA for:

- analytical sampling, sample shipments, chain-of-custody documentation and laboratory QA, and
- radiological measurement data reduction, data evaluation, and reporting.

QA records will also be maintained for the following activities: operating logs, results of reviews, inspections, tests, audits, work performance monitoring, and materials analyses. The records will also include data such as qualifications of personnel, procedures, and equipment. Testing and inspection records will identify the inspector or data recorder, the type of observation, the results, the acceptability, and any actions taken regarding deficiencies noted. Operational records will be identifiable and retrievable, and be retained by the RSO until licenser termination at the mine site.

Management of the QA Program (EPA, 2002) will take into account the following applicable items.

- Data Quality Objectives (DQO) for a specific monitoring program, as noted in EPA (2006), that provides development examples to define acceptance and performance criteria.

- A Quality Assurance Project Plan (QAPP) that documents data collection, analysis, assessment, and how to achieve expected data quality.

5.7.9.1 Organizational Structure and Responsibilities of Managerial and Operational Personnel

The following QA program elements will be developed and implemented to ensure the quality of data/results for radiological effluent and environmental monitoring programs.

The QA program will address the following topics:

- management and operation of the monitoring programs;
- QA policy and functions, definition and documentation; and
- authorities, duties, and responsibilities of the organizational positions.

LC ISR, LLC personnel or contracted organizations conducting QA functions will be given sufficient authority and organizational freedom: to identify quality problems; to initiate, recommend, or provide solutions; and to verify implementation of solutions. Reporting will be at the managerial level, independent of activity performance, costs, and schedule. Applicable organizational structure guidance from Section 2.1.1 of ANSI/ ASQC (1994) and Section 5.2.1 of ANSI (2003) will be reviewed regarding the organizational responsibilities for radiological effluent and environmental monitoring programs.

5.7.9.2 Specification of Qualifications of Personnel

Qualified LC ISR, LLC personnel will carry out assigned radiological monitoring functions per their background and job description. Qualified and responsible individuals will be trained in the principles and techniques of the activities to be performed, including maintaining proficiency by retraining, reexamining, and recertifying or by periodic performance reviews, as appropriate. Applicable guidance and criteria in Section 2.3.1 of ANSI/ASQC (1994) and on page seven of NR RG 4.15 will be evaluated regarding personnel development, training and qualification specifications.

5.7.9.3 SOPs and Instructions

Monitoring programs should have written procedures for all activities that generate data, such as dose calculations and measurements, sample collection, sample management and chain-of-custody documentation, sample preparation and analysis, data reduction and recording, data assessment and reporting, and final sample disposal. Procedures are also needed for addressing support functions (such as operation of process monitors, training, and preparation of QC samples) and collection of meteorological data, corrective actions, audits, and records. Qualified individuals will participate in preparation, review, and revision of these procedures.

5.7.9.4 Records

LC ISR, LLC will maintain a system that produces unequivocal, accurate records that document all monitoring activities, such as:

- procedure revision;
- personnel training and qualification records;
- analytical results;
- audits;
- corrective actions;
- intermediate activities or calculations (as may be needed to validate or substantiate final results);
- records of tracking and control (e.g., chain-of-custody documentation) throughout all processes from sample collection;
- thorough analysis and reporting of results, including unique identifiers, descriptions, and sources;
- dates/times, packaging/preparation/shipping, and required analyses;
- field logs and records with sufficient information that describe environmental conditions;
- information and data that document the nature of the sample and where and how it was taken;
- electronic data collection and algorithms and QA documentation;
- calculations (including data reduction, analysis, and verification) NRC RG 4.15, page eight;
- QC records for radiation monitoring equipment, including the results of the radioactive source;
- checks, calibrations, instrument background determinations, and maintenance activities affecting equipment performance; and
- notifications to qualified staff of any approved procedural changes affecting data quality.

Records will be legible and identifiable, retained at the mine site, and will be protected against damage, deterioration, or loss. These records will be maintained in a format that is easily retrievable, physically or electronically, and can be reviewed in an uncorrupted form.

LC ISR, LLC will use the appropriate guidance from applicable sections in ANSI/ASQC (1994); ASME (1994) ; Section 4.13 of ISO/IEC (2006); and TG11-1998, TG15-1998, TG16-1998, and TG21-1998 (NIRMA, 1998a, b, c, and d).

5.7.10 Respirators

The RSO will develop and administer a respirator program consistent with the guidance found in NRC RG 8.15 and regulations found in 10 CFR 20.1703. Respirators will only be worn when administrative and engineering controls are insufficient to provide adequate protection from airborne particulate. Since the Plant will not have a yellowcake drying and packaging facility, the opportunities for exposure to airborne particulate are generally limited to maintenance and upset conditions. The respirator program will address the following issues.

- Only National Institute for Occupational Safety and Health (NIOSH)-approved respirators will be used.
- An air sampling program will be sufficient to identify potential hazards, permit proper equipment selection, and assign doses.
- Surveys and bioassays will be performed as necessary to evaluate actual intakes.
- Respirators will be tested for operability (including fit testing).
- Written procedures will describe proper respirator use and maintenance.
- Medical evaluations will demonstrate the employee is capable of using a respirator.

The RSO will evaluate the effectiveness of existing engineering and administrative controls in order to determine when a respirator is required. Due to the physiological and psychological stresses imposed by respirators, they will only be used as a last line of defense against airborne particulate.

5.8 Transport of Radioactive Materials

The Lost Creek ISR Project will produce wet yellowcake slurry that will be sent to another facility for final drying and packaging. Also, Lost Creek ISR, LLC has designed the Lost Creek Plant to receive and process resin and/or slurry from other facilities for toll processing. The following two sections describe: the risks associated with the transport of radioactive material as well as the associated mitigations that will be utilized to control those risks; and how Department of Transportation rules and regulations pertaining to the transport of radioactive materials will be implemented.

5.8.1 Risks of Transporting Radioactive Material

NUREG/CR-6733 documents a transportation hazard risk analysis in Section 4.5. While this analysis was for largely for dry yellowcake and not for slurry or resin that will be transported from (slurry) or to (slurry and/or resin) the Lost Creek Project, the analysis is

still relevant since the risks posed by dry yellowcake are generally more significant than those posed by wet slurry or resin. In fact, NUREG/CR-6733 briefly addresses the differences between dry yellowcake and slurry and resin at the end of Section 4.5.1.

The NUREG analysis considered the distances travelled to a processing facility in Illinois and a specific number of trips per year. The NUREG analysis shows that the probability of a truck accident in any given year for a facility producing 1.3 million pounds of yellowcake per year is 11%. The analysis also considered the potential for exposure to a member of the public as a result of an accident. Two separate exposure models were developed based on the amount of material released. Model I assumed a spill of 45% of the load and resulted in a potential exposure of 200 person-rem. Model II assumed a spill of 3% of the contents and resulted in a potential exposure of 14 person-rem.

The likelihood of an accident involving a shipment of resin or slurry, when considered on the basis of distance transported, is the same as that for dried yellowcake. However, the consequences of a spill of resin or slurry should be analyzed differently since they present different radiological risks and potentials for release of material. Also, the number of miles driven per year may be significantly less than that considered by the NUREG. The analysis performed for resin and slurry below will consider the distance driven per year.

5.8.1.1 Resin

Resin will be transported in a tanker vessel that complies with DOT requirements for design, testing, and placarding. The resin is typically composed of styrene copolymerized with divinylbenzene (DVB). The resin presents no chemical hazard itself. The resin beads are extremely slick when walked due to their round shape. The resin contains a large fraction of water and is therefore not combustible until all of the water has been evaporated. The resin will typically contain around 7 pounds of uranyl carbonate per cubic foot of resin. Therefore, a 500 cubic foot tanker load of resin will contain on the order of 3,500 pounds of uranyl carbonate (compared to a load of dry yellowcake which typically contains on the order of 33,000 pounds of uranyl peroxide).

The uranyl carbonate is bonded tightly to the resin beads. Since the resin beads are not combustible there is only one way to remove the uranyl peroxide from the resin and that is through elution with a concentrated salt such as sodium chloride or with a strong acid such as hydrochloric acid. It is extremely unlikely that a spill of resin would come into contact with a concentrated salt solution or a strong acid.

Inhalation or ingestion of the resin is also extremely unlikely given that their density is slightly greater than water and resin does not readily become airborne. As stated previously, the resin is designed to capture uranyl carbonate. Due to its electrochemical

charge the resin will not capture any of the uranium daughter products. However, small quantities of daughter products could be physically attached to the resin or could grow in due to radioactive decay but the daughters will not be chemically attached to the resin. The water used to handle resin at a processing plant is removed before shipping and in many cases the resin is rinsed with fresh water to minimize the presence of contaminants such as radium. The only credible radiological risk associated with resin would be from direct radiation from uranium and its ingrown daughters. Since uranium is an alpha producer and there is no opportunity for inhalation or ingestion and alpha particles cannot penetrate the skin, there is no opportunity for exposure from the uranium. During production, once resin is loaded with uranyl carbonate it is typically shipped within one to three days to the processing facility. This short time span will give very little opportunity for daughters to grow in. Therefore, the risk from ingrown daughters is minimal.

The density of loaded ion exchange resin is slightly greater than water (1.00 mg/cm^3 for pure water and 1.09 mg/cm^3 for typical ion exchange resin loaded with 7 pounds of uranyl peroxide per cubic foot). The similar densities will allow resin to quickly dilute if it is spilled into a body of water.

Typically resin can only be economically transported very short distances because each shipment contains so few pounds of uranium. Based on the location of other in situ facilities in the region of the Lost Creek Project and the economics of shipping, it is unlikely that any company would ship resin over 100 miles for processing. While this single trip mileage is significantly less then that analyzed by the NUREG, the total number of trips per year would climb to around one trip per day. Therefore, if it is assumed the average trip is 50 miles per day the yearly mileage would be 18,250 miles. Using an accident rate presented in the NUREG of 1.4×10^{-6}/km of rural two-lane road, the likelihood of an accident in any given year would be 4%. The risk of returning the barren resin back to Lost Creek was not considered since the radiological risk is negligible.

5.8.1.2 Slurry

Slurry will be transported in a tanker vessel that complies with DOT requirements for design, testing, and placarding. Slurry is simply yellowcake mixed with water (typically 40% to 60% water by weight). If a slurry spill is cleaned up quickly, it does not present an air borne hazard like dry yellowcake. Shipments of slurry contain around one half the amount of uranium that a shipment of dry yellowcake contains because of the extra water weight that makes up the shipment of slurry.

It is unknown what the likelihood of spilling product from a tanker trailer is compared to spilling drummed material.

Slurry can be shipped economically a considerable distance. It is unknown at this time where slurry from the Lost Creek Project will be dried and packaged but for this risk assessment a conservative distance of 1,300 is considered. A total of 200 miles is assessed as urban interstate with an accident rate of 1.4 x 10^{-6}/km and 1100 miles is considered rural interstate with an accident rate of 4.0 x 10^{-7}/km. A facility producing 1,000,000 of yellowcake per year will need to ship slurry a total of 63 times per year. These rates yield an accident likelihood of 7% per year. The risk of returning the empty vessel back to Lost Creek was not considered since the radiological risk is negligible.

5.8.2 Prevention and Mitigation of Transportation Accidents

The prevention and mitigation of transportation accidents involving radioactive materials is of the utmost importance. Toward that end, the writing and implementation of a robust program, which complies with all applicable Department of Transportation regulations, will be overseen by the EHS Supervisor/RSO. The tenants of the program will include but will not be limited to the following:

- DOT Hazmat training for all employees who are involved in security, select packaging containers, package, label, placard, complete shipping papers, respond to emergencies or otherwise deal with shipments of radioactive materials;
- Proper identification of hazard class;
- Proper selection of placarding and positioning on the trailer;
- Proper container labeling;
- Proper packaging;
- Completion of shipping papers;
- Proper radiological survey and release of shipments as well as empty containers;
- Proper positioning of a load;
- Proper binding of load (blocking, bracing, etc.)
- Qualifications of hazmat drivers;
- Security (including but not limited to storage of unloaded and loaded materials and confirmation of driver identity);
- Response to emergencies (clean-up of spills, training of emergency responders, qualification of responders, establishment of response kits); and
- Pre-shipping inspections to ensure proper packaging, labeling, placarding, and shipping papers.

Additionally, the EHS Supervisor/RSO will ensure the program considers and complies with all applicable provisions of the following regulations:

- 10 CFR 71
- 49 CFR 107, subpart G: Hazardous material shipper/carrier registration
- 49 CFR 171 and 171.16: Accident reporting
- 49 CFR 172, subpart C and G: Shipping papers and emergency information
- 49 CFR 172, subparts D and E: Marking and labeling
- 49 CFR 172, subpart F and appendixes B and C: Placarding
- 49 CFR 172, subpart H: Hazardous material employee training
- 49 CFR Part 173, subparts A, B, and I: Packaging
- 49 CFR Part 177 and 390-397: Specifically governing transport on public highways

No shipments of radioactive material will occur until the entire program is written and functioning.