Review and Approvals	
Prepared: Jay Maisler	
Signature:	Date:
Reviewed by Dane Watson	
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Reviewed by Quality Assurance Coordinator: Chuck Beatty	
Signature:	Date:
Approved by Radiation Safety Officer: Jay Maisler, CHP	
Signature:	Date:
Approved by Trustee Project Manager: Jeff Lux	
Signature:	Date:
Approved by Administrator, Cimarron Environmental Response Trust: Bill Halliburton	
Signature:	Date:

NOTE

The content of Revision 5 to the Radiation Protection Plan (RPP) includes provisions that apply to activities proposed in the *Cimarron Environmental Respose Trust Facility Decommissioning Plan* (D-Plan) that is under review by the U.S. Nuclear Regulatory Commission (NRC). This RPP is incorporated as Appendix M of that D-Plan. Upon approval by the NRC, this RPP will be revised to incorporate any changes required by the NRC staff during the review process, as documented in a Safety Evaluation Report. RPP provisions that cannot be implemented without NRC approval are identified in GRAY HIGHLIGHTED TEXT. Other changes in Rev. 5 have been reviewed in accordance with license condition 27(e) of SNM-928 by the ALARA Committee and determined that they may be implemented as they do not:

- Conflict with the ALARA principle or decommissioning process.
- Conflict with requirements specifically stated in the license or impair the Cimarron Environmental Response Trust's ability to meet all applicable NRC regulations.
- Cause degradation in safety or environmental commitments addressed in the NRC-approved RPP and/or D-Plan or have a significant adverse effect on the quality of the work, the remediation objectives, or health and safety.
- Conflict with the conclusions analyzed in the Environmental Assessment, dated July 29,1999, and Safety Evaluation Report, dated August 20,1999.

Summary of Changes

Revisions to this document will be identified, and revisions or addenda will be issued as needed. The end user is responsible to verify that any hard copy being used is the current revision. A current version of the RPP is maintained on the Cimarron SharePoint site. A hard copy is available at the Site Office. A summary description of each revision or addendum will be noted in the following table.

Revision Number	Date	Comments
Rev. 0	April 11, 2011	Original
Rev. 1	Feb. 3, 2012	Revision 1 to the Cimarron RPP contains numerous administrative changes and editorial changes. Specific changes are identified in a separate 27(e) evaluation and attached markup of changes from RPP Rev. 0.
Rev. 2	Feb. 24, 2014	Revision 2 to the RPP includes clarifications addressing groundwater processing and editorial changes. Specific changes are identified in a separate 27(e) evaluation and attached markup of changes from Rev. 1.
Rev. 3	April 15, 2016	Revision 3 provides changes to support the proposed Decommissioning Plan and includes editorial changes. Clarifications were added to address how radiological controls for routine activities are handled when an activity plan is not required or used. Specific changes are identified in a separate 27(e) evaluation and attached markup of changes from Rev. 2.

Revision Number	Date	Comments
Rev. 3.1	Dec. 31, 2018	Revision 3.1 is an interim revision This is an interim revision pending approval of draft Revision 4 by the NRC. Accordingly, it is labeled Revision 3.1. This revision to the RPP incorporates changes identified during the submittal of the Cimarron Facility Decommissioning Plan, Rev. 1, that the ALARA Committee determined can be implemented prior to NRC approval of the Decommissioning Plan. This revision also addresses and clarifies issues discussed with the NRC staff during the November 2018 inspection at the Site.
Rev. 3.2	Sep. 15, 2019	Includes editorial corrections. Corrected error in Section 8.2 that implied personnel access logs were required for entry into areas posted solely as Radioactive Material Areas. Corrected typographical area regarding Radioactive Materials Area in Table 8.1 and Section 10.2.

Revision Number	Date	Comments
Rev. 4	March 1, 2021	Added signature blocks to Reviews and Approvals page to identify preparer and technical reviewer. Changed language on verification of the current version of the RPP in accordance with Site practices in Summary of Changes and footers. Includes changes made to conform with the Cimarron Site Decommissioning Plan under review by the NRC for approval Added a Note on the cover page to indicate that certain changes identified within the RPP cannot be implemented until NRC approves the Decommissioning Plan. After NRC approval, Rev. 5 to the RPP will be issued to reflect NRC approved RPP language. Other changes have been to clarify Site implementation of the RPP or to make editorial corrections and clarifications. Specific changes approved for implementation are identified in a separate 27(e) evaluation and attached markup of changes from Rev. 3.2.
Rev. 5	Oct. 1, 2022	Editorial and other minor corrections. Additional information added in response to NRC Staff Request for Supplemental Information and Clarification (9/2021) and questions from the pre-submittal inspection (1/2022). Conforming changes were made to ensure consistency with the Decommissioning Plan Rev. 3.

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1.0 INTRODUCTION

1.1 Purpose

This Radiation Protection Plan (RPP) establishes radiation protection requirements implemented at the Cimarron Site to achieve compliance with applicable regulatory requirements and License SNM-928. As provided in the *Cimarron Environmental Response Trust Facility Decommissioning Plan,* Revision 3 (D-Plan), the RPP will be implemented during decommissioning (extraction and treatment of uranium-impacted groundwater).

1.2 Scope

The RPP applies to all radiological operations, routine and emergency, at the Cimarron Site. The RPP applies to the following personnel when present at the Cimarron Site:

- Licensee employees
- Contractors and their employees
- Visitors

2.0 TRAINING REQUIREMENTS AND POLICY

2.1 Section Overview

This section describes radiation safety training requirements for individuals who enter a Restricted Area (RA), handle radioactive material, or work in the vicinity of radioactive material at the Site.

2.2 Responsibilities

The RSO is responsible for the radiation safety training program which includes:

- Approving radiation safety training materials
- Approving personnel performing radiation safety training
- Performing radiation safety training
- Verifying that those individuals who require radiation safety training receive appropriate training

2.3 Training Requirements

Radiation safety training requirements are tiered to provide an appropriate level of training based on the potential for radiation exposure of an individual at the Cimarron Site. Individuals who visit or work at the Site but do not require unescorted access to RAs or Radioactive Material Areas must complete Radiological Orientation. In addition to Radiological Orientation, General Worker Radiological Training is required for individuals requiring unescorted access to RAs or Radioactive Material Areas whose duties do not involve working with or handling radioactive material. Radiation Worker Training is required for individuals who handle or work directly with radioactive materials.

The Licensee shall not assume that radiation safety training has been adequately covered by prior employment or academic training.

Inspectors and representatives of the NRC and the Oklahoma Department of Environmental Quality, Land Protection Division, Radiation Management Section are exempt from radiation safety training. Site-specific information may be provided to agency personnel if deemed necessary by the RSO.

Prospective evaluations of radiological conditions and potential doses to workers for the groundwater treatment process have been performed (see Appendices A and B). Based on the results of these evaluations, there is no need for individual monitoring. General Worker Radiological Training (Subsection 2.3.2) or Radiation Worker Training (Subsection 2.3.3) for workers accessing RAs, and the boundaries of any required RA(s) is required as discussed below.

2.3.1 Radiological Orientation

Radiological Orientation is required for visitors and individuals visiting or working at the Cimarron Site but not permitted to enter RAs or Radioactive Material Areas. Individuals who complete Radiological Orientation will be granted escorted (i.e., under direct supervision of a Qualified Escort) access to RAs and Radioactive Material Areas but working with or handling radioactive materials is not permitted. Entry into Contaminated Areas, Airborne Radioactive Material Areas, or areas requiring either bioassay or respiratory protection is not permitted. Activities these individuals undertake may include, but is not limited to general office work, housekeeping, and tours and inspections of the property.

Information required for Radiological Orientation may be presented in a classroom setting or provided as a "read-and-sign" document. A test is not required for Radiological Orientation. Documentation will be maintained for all individuals completing Radiological Orientation. The following topics will be addressed:

- Radioactive materials that are present at the Site
- NRC Form 3, "Notice to Employees"
- Information regarding radiation safety requirements for work to be performed (e.g., groundwater sampling, well installation, groundwater processing, packaging and shipping for disposal, etc.)
- Site access and egress (typically covered in Site Safety & Health Orientation);
- Response to emergency conditions (including weather, fires, personnel injuries) (typically covered in Site Safety & Health Orientation);
- Site industrial safety requirements (typically covered in Site Safety & Health Orientation)

Refresher training for Radiological Orientation shall be conducted annually (within 12 months).

2.3.2 General Worker Radiological Training

General Worker Radiological Training is required for workers who are granted unescorted access to RAs and Radioactive Material Areas but who are not permitted to:

- Work with or handle radioactive material.
- Enter Contaminated Areas or Airborne Radioactive Material Areas.
- Enter areas where bioassay or respiratory protection is required.

Information required for General Worker Radiological Training may be presented in a classroom or virtual classroom setting or provided as a "read-and-sign" document. Documentation will be maintained for all individuals completing General Worker Radiological Training. General Worker Radiological Training will include:

- Information covered in Radiological Orientation described above;
- Information regarding the principles and practices of radiation protection;
- Information regarding the purpose and functions of protective and monitoring devices that will be used, as applicable;
- Information regarding protection available for the embryo/fetus, as applicable.

General Worker Radiological Training will include a test to verify an adequate understanding of the training. Each test shall have a minimum passing grade of 80%. If an individual does not pass the test, the test may be administered a second time. If the candidate fails the test a second time, the candidate must repeat the entire General Worker Radiological Training course before he/she can take another test.

A candidate who does not achieve a minimum of 80% on the General Worker Radiological Training test will not be granted unescorted access to any RA or Radioactive Material Area.

Refresher training for General Worker Radiological Training shall be conducted annually (within 12 months) and does not require re-testing.

2.3.3 Radiation Worker Training

Radiation Worker Training is required for individuals who in the course of employment are likely to receive an occupational dose to radiation greater than 100 mrem (1

millisievert) in a year or whose duties require them to work in an RA, Radioactive Material Area, Contaminated Area, Airborne Radioactive Material Area, or routinely work with or handle radioactive material, or use respiratory protection equipment (for radiation protection). Such workers may include groundwater processing operators and their supervision, health physics technicians, and environmental sampling personnel. Individuals successfully completing Radiation Worker Training are designated as Qualified Escorts.

Radiation Worker training will include:

- Information covered in General Worker Radiological Training described above;
- Radioactivity measurements, monitoring techniques, and usage of monitoring instrumentation;
- Basic calculations involved in using and measuring radioactivity;
- Types of radiation, range and effects;
- Regulatory and Site-specific dose limits to the general public and occupationally exposed persons;
- Storage, transfer, or use of radiation and/or radioactive material;
- Biological effects of radiation;
- Health protection problems associated with exposure to radiation and/or radioactive material;
- Precautions or procedures to minimize exposure;
- Purposes and functions of protective devices employed;
- Applicable NRC regulations and license requirements for the protection of personnel from exposure to radiation and/or radioactive material including

responsibility to observe regulatory and license requirements to the extent within the worker's control;

- Workers' responsibility to report promptly to the Licensee any condition which may lead to or cause a violation of Commission regulations and licenses or unnecessary exposure to radiation and/or radioactive material;
- Appropriate response to warnings made in the event of any unusual occurrence or malfunction that may involve exposure to radiation and/or radioactive material;
- Radiation exposure reports which workers may request pursuant to 10 CFR 19.13.

Initial Radiation Worker Training will include a test to verify an adequate understanding of the training. Each test shall have a minimum passing grade of 80%. Each test question answered incorrectly shall be reviewed with the test participant and noted on test. If an individual does not pass the test, the test may be administered a second time. If the candidate fails the test a second time, the candidate must repeat the entire Radiation Worker Training course before he/she can take another test.

A candidate who does not achieve a minimum of 80% on the Radiation Worker training test will not be permitted to perform work in an RA or to handle radioactive material until such time as the Radiation Worker training and test is successfully completed. The individual may continue unescorted access to RAs or Radioactive Material Areas if they successfully completed General Worker Radiological Training.

2.3.4 Training Delivery

Any of the following techniques, or combination thereof, may be used for radiation safety training:

- Classroom training

- Audiovisual media
- Reading assignments (Self Study)
- Computer-based or on-Line training (Internet)

On-the-job training (OJT) under the presence of an individual trained in the specific activity being observed;

- Demonstrations
- Drills
- Discussions

2.3.5 Training Frequency

- Initial Radiation Worker Training shall be conducted before working an RA, Radioactive Material Area, Contaminated Area, Airborne Radioactive Material Area, or routinely in working with or handling radioactive material.
- A training update for Radiological Orientation, General Worker Radiological Training, or Radiation Worker Training shall be provided, as appropriate, whenever there is a significant change in duties, regulations, or terms of the license.
- Refresher for Radiation Worker Training shall be conducted annually (within 12 months) and does not require re-testing.

2.4 Refresher Training

Refresher training for Radiological Orientation, General Worker Radiological Training, and Radiation Worker may be satisfied by the RSO or designee issuing required reading that is formally acknowledged by the individual in an email or signed acknowledgement form. Other

methods for conducting refresher training may be required by the RSO based upon lessons learned throughout the year.

2.5 Training Records

Training records shall include the following documentation:

- Rosters of individuals attending Radiological Orientation briefings.
- Rosters of individuals attending General Worker Radiological Training.
- Rosters of individuals attending Radiation Worker Training.
- Completed General Worker Radiological Training graded tests.
- Completed Radiation Worker Training graded tests.
- Documentation of completion of refresher for General Worker Radiological Training.
- Documentation of completion of refresher for Radiation Worker Training.

Records for all individuals shall be maintained in accordance with the Quality Assurance Program Plan (QAPP).

3.0 ADMINISTRATION AND RESPONSIBILITIES

3.1 Section Overview

This section describes the radiation protection organization and responsibilities of those individuals implementing the RPP.

Administration of the radiation protection program requires coordination among the following individuals:

- Trust Administrator
- Trustee Project Manager (Trustee PM)
- RSO
- Quality Assurance Coordinator (QAC)
- Task Specific Project Managers (PMs)
- Activity Leaders
- Individual Workers
- ALARA Committee

3.2 Radiation Protection Organization

The radiation protection organizational structure for the Cimarron Site is shown in Figure 3-1.

Trust Administrator – The Trust Administrator is responsible for the management of Trust assets and provides the resources needed to complete the decommissioning of the site. The Trust Administrator monitors and reports the financial status of the Trust accounts. The Trust Administrator is responsible for the preparation of periodic decommissioning funding cost estimates and annual budgets. The Trust Administrator is a permanent member of the Site ALARA Committee.

The Trust Administrator must have experience managing organizations responsible for radiological decommissioning and environmental remediation, as well as overseeing the preparation of financial reports and cost estimates.

Trustee PM – The Trustee PM is responsible for overseeing the construction and operation of decommissioning systems, the implementation of radiation safety, industrial health and safety, quality assurance, and environmental compliance programs. The Trustee PM is responsible for ensuring that all personnel performing decommissioning activities, or working in radiation protection, health and safety, quality assurance, or environmental compliance functions receive training and have the skills and experience required to perform those functions. In conjunction with the Trust Administrator, the Trustee PM prepares decommissioning cost estimates and annual budgets. The Trustee PM retains contractors/consultants with appropriate qualifications and experience to maintain and implement radiation protection, quality assurance, and safety & health programs. The Trustee PM is a permanent member of the ALARA Committee.

The Trustee PM reports directly to the Trust Administrator. The Trustee PM must have experience in the following areas:

- Managing environmental remediation and/or radiological decommissioning projects
- Complying with license and regulatory requirements
- Preparing and tracking work scopes and cost and schedule plans

RSO (Jay Maisler) – The RSO is responsible for maintenance and implementation of the radiation protection program. The RSO is also responsible for review and revision of the RPP and procedures, radiation exposure monitoring, dose reporting, the radiological instrument program, and all levels of radiation safety training. The RSO is responsible for ensuring that all activities comply with license requirements, chair the ALARA Committee, and manage the

health physics staff. The RSO is given specific authority to implement and manage the Licensee's radiation protection program, either directly or through qualified individuals who are designated in writing as having authority to exercise specific functions. All radiation protection personnel have stop work authority.

The responsibility for the implementation and review of the Material Control and Accountability(MC&A) program is assigned to the RSO for the Cimarron Site. The RSO establishes training programs and qualifies those individuals to whom the RSO delegates responsibility to implement the (MC&A) Plan.

The RSO reports directly to the Trustee PM, but also has a direct communication line to the Trust Administrator. The RSO must have the following qualifications:

- Knowledgeable of potential radiological hazards and emergency preparedness associated with decommissioning activities at the Cimarron Site
- Completed educational courses related to ionizing radiation safety, or a radiation safety officer course, or maintains designation as a Certified Health Physicist
- Experience managing and implementing radiation protection programs at decontamination and decommissioning sites
- Background in license compliance
- Familiarity with license and regulatory requirements
- Familiarity with site-specific radiation protection, quality assurance, health and safety, and sampling and analysis programs
- Experience in performing ALARA evaluations
- Overseeing radiological characterization and final status surveys

Experience in decontamination and decommissioning projects

QAC – The QAC is responsible for the maintenance and implementation of the quality assurance program. The QAC performs or schedules periodic and/or ad hoc audits and observations of all decommissioning and program management functions. All quality assurance personnel have stop work authority. The QAC is also responsible to perform periodic evaluations of the effectiveness of the quality assurance program and to ensure that all personnel performing quality-critical tasks have received the appropriate level of training on the Site-specific quality assurance program. The QAC is a standing member of the Site ALARA Committee.

The QAC reports to the Trustee PM, but also has a direct communication line to the Trust Administrator. The QAC is required to have the following qualifications:

- Experience in managing quality control/quality assurance programs
- Familiarity with license and regulatory requirements
- Familiarity with Site-specific radiation protection, quality assurance, health and safety, and sampling and analysis programs
- Familiarity with data verification and validation protocols

PMs – PMs are responsible for the preparation of plans, procurement of services and materials, and the execution of decommissioning projects. PMs ensure that all personnel working on projects have received all the training needed and are qualified to perform the tasks for which they are responsible to perform. PMs are responsible for monitoring the schedule, cost, and quality of the project work.

Activity Leader and Front-Line Supervisor – Activity Leaders (ALs) are the front-line supervisors over non-routine work performed at the Cimarron Site. ALs are responsible for the

preparation of activity plans and procurement of services and materials for non-routine activities. Front-line supervisory personnel are responsible for procurement of services and materials and the performance of decommissioning operations for routine operations. ALs and front-line supervisors ensure that all personnel working on projects are familiar with the activity plan under which the work is being performed, and that they have received all the training needed and are qualified to perform the tasks for which they are responsible to perform. ALs and front-line supervisors are responsible for monitoring the schedule, cost, and quality of the project work.

Individual Worker – Each worker is responsible for their own protection and the protection of their co-workers. Workers should know how NRC requirements relate to their work and should follow them. If a worker observes violations of the requirements or has a safety concern, they should report them as discussed in section 3.3.2 of this Plan. Workers are provided training related to their responsibilities in Radiological Orientation, General Worker Radiological Training, and Radiation Worker Training.

ALARA Committee – An ALARA Committee has been established in accordance with regulatory and license requirements. Throughout the decommissioning of the Site, the ALARA Committee is responsible to ensure that procedures and engineering controls used are based upon sound radiation protection principles to achieve occupational doses and dose to members of the public that are ALARA. The ALARA Committee will meet at least once per quarter. The responsibilities of the ALARA Committee are discussed in Section 4.3.1.

3.3 Policies

3.3.1 Stop Work Authority

All Site personnel have the authority to stop work:

- If radiological health and safety of workers is compromised
- If radiological health and safety of the general public is compromised
- If radiological regulatory non-compliance may occur (includes NRC regulations, license conditions, and radiation protection procedures)

3.3.2 Reporting Safety Concerns and Regulatory Violations

All workers at the Site have the right to report safety concerns and observations of potential regulatory or license violations. Individuals are encouraged to contact the RSO first if they have a radiation safety concern or observe a potential regulatory or license violation. This is not a requirement.

Individuals who are not satisfied with the response to an expressed concern have the right to contact the NRC for resolution. See NRC Form 3, "Notice to Employees." No penalty or retribution will result to an individual who contacts the NRC.

3.4 Radiation Protection Program Document Hierarchy

The order of precedence in regulating the Cimarron Site is:

1. Federal radiation protection regulations (10 CFR)
2. License SNM-928, including the RPP which is incorporated into the license via a license condition
3. Radiation protection program procedures

3.5 Procedure Development

Radiation protection procedures have been developed to provide consistent, effective performance of radiation protection activities. Radiation protection procedures include, but are

not limited to, the sampling and analysis of influents and effluents to monitor the accumulation of special nuclear material in resins, the sampling of loaded resin for waste characterization, and the sampling, analysis, handling, storage, manifesting, transportation, and disposal of low-level radioactive waste.

Radiation protection procedures shall be developed in accordance with the QAPP.

Radiation protection procedures shall comply with regulatory requirements, license conditions, and the RPP.

Radiation protection procedures may incorporate or reference applicable technical guidance documents (e.g., NRC Regulatory Guides and NUREGs, National Council on Radiation Protection and Measurements (NCRP) guidance, International Council on Radiation Protection (ICRP) guidance, American National Standards Institute (ANSI) documents, etc.).

3.6 Procedure Review, Approval, and Control

Radiation protection procedures shall undergo technical verification and review to ensure compliance with regulatory requirements, applicable licenses and permits, and the RPP, as well as conformance, to the extent practicable, with applicable industry standard practices.

- Radiation protection procedure review shall assess compatibility with all other Licensee plans, manuals, and procedures.
- Radiation protection procedure review shall ensure that the procedure can be performed as written.
- All radiation protection procedures shall be reviewed and approved by the RSO.
- All radiation protection procedures shall by reviewed by the QAC or designee for conformance with quality assurance program requirements.

All radiation protection procedures shall be controlled in accordance with the QAPP.

3.7 Desk Instructions

Desk Instructions may be developed and implemented to provide guidance on radiation protection program implementation or to clarify program implementation expectations from the RSO. Desk Instructions serve as a reference guide on specific topics that help the user implement various aspects of the RPP. Desk Instructions may be written to provide instructions for performing routine or special radiological surveys, qualify or requalify individuals to perform radiological surveys, or identify RSO designees. Desk Instructions are issued by the RSO or designee and expire 12 months after approval. Desk Instructions may be renewed at additional 12-month increments. All Desk Instructions shall be controlled in accordance with the QAPP.

3.8 Notifications and Reports

Notifications and reports shall be made in accordance with the requirements of 10 CFR 19, 10 CFR 20 and 10 CFR 70. Detailed instructions for regulatory requirements related to notifications and reports are provided in radiation protection procedure, RP-05, "Radiation Protection Reports and Assessments."

3.8.1 Required Notices and Postings

The RSO responsible for ensuring the following postings and reports available to employees and contractors working at the Site. In some cases, where the volume of pages associated with a required posting or report is impractical to physically post, notice informing workers where the information is available or how it can be obtained may be posted:

- Current NRC Form 3, "Notice to Employees"

- 10 CFR 19 and 10 CFR 20 regulations
- A copy of SNM-928 and documents incorporated by license, reference, and amendments to the license.
- Operating procedures applicable to licensed activities.
- Any notice of violation involving radiological working conditions, proposed imposition of civil penalty, or order issued by the NRC, and any response from the Trust.

3.9 RSO Designees and Qualifications

The RSO shall document the qualifications of individuals designated to perform specific RSO responsibilities ("designee"). Prior to designating an individual, the RSO considers the following and documents the results of these considerations for each designee:

3.9.1 RSO Designees

The designated individual should have a bachelors' degree in the physical sciences, industrial hygiene or engineering from an accredited college or university. If the designated individual does not meet the educational guideline, then the individual may have the equivalent combination of training and relevant experience in radiological protection. Two years of relevant experience are considered equivalent to 1 year of academic study (e.g., in lieu of a bachelors' degree, the individual should have a combination of eight years relevant training and experience).

The designated individual should have at least one year of work experience in applied health physics, industrial hygiene or similar work relevant to radiological hazards associated with site remediation. This experience should involve actually working with

radiation detection and measurement equipment, not simply administrative or "desk" work.

The designated individual should have a thorough knowledge of the proper application and use of all health physics equipment used for the radionuclides present at the Site, the chemical and analytical procedures used for radiological sampling and monitoring, and methodologies used to calculate personnel exposure to the radionuclides present at the Site. The individual must have the appropriate specialized knowledge to perform the designated responsibility.

3.9.2 Health Physics Technicians

The designated individual should have a bachelors' degree in the physical sciences, industrial hygiene or engineering from an accredited college or university. If the designated individual does not meet the educational guideline, then the individual may have the equivalent combination of training and relevant experience in radiological protection. Two years of relevant experience are considered equivalent to one year of academic study (e.g., in lieu of a Bachelors' degree, the individual should have eight years relevant experience). This experience should involve actually working with radiation detection and measurement equipment, not simply administrative or "desk" work.

3.9.3 Task Qualified Individuals

Designated individuals may be qualified to perform specific tasks approved by the RSO (e.g., "task qualification"). A modified "systematic approach to training" is employed to qualify individuals on specific tasks. Task qualifications must be documented and shall include the following:

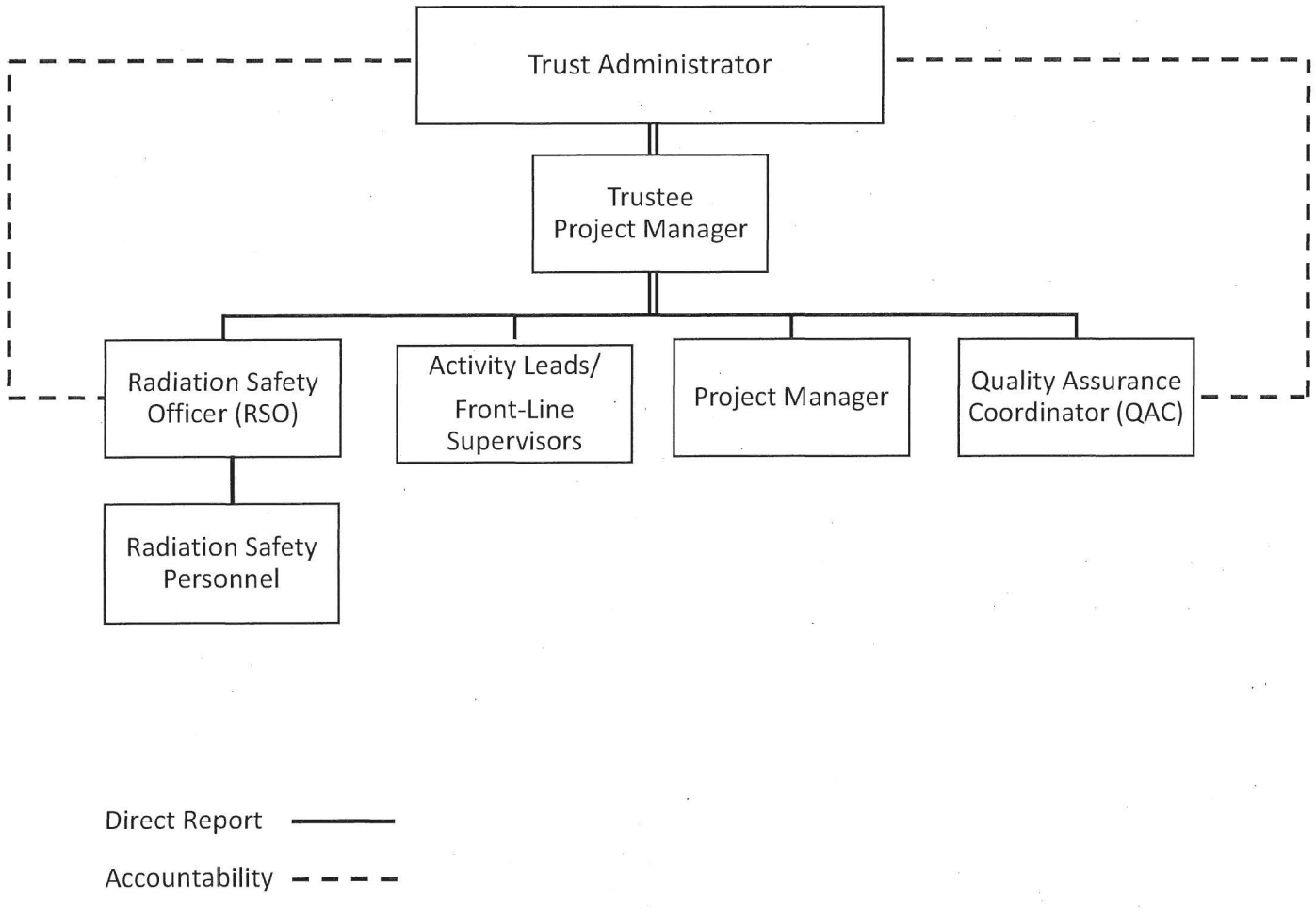
- Verification that the selected individual has sufficient experience (e.g., related technical experience, such as environmental remediation, industrial hygiene, use of scientific instruments, etc.), education (including physical science and math), and prior training (related to the specific task, which may include electronic equipment use and handling, computer applications, etc.).
- Learning objectives based on the procedural requirements to perform the task.
- On-the-job training including performance terminal objectives that the individual must satisfy through performance, simulation, or discussion. Each performance terminal objective should include the behavior being evaluated (e.g., task being performed), conditions associated with the task, standards that must be met (e.g., applicable procedures), and the steps necessary to perform the specific task.

3.9.4 Expiration of Task Qualifications.

Task qualifications are typically valid for 12 months. Refresher training, including review of instrument use, survey documentation, and procedure updates over the past year shall be provided at least every 12 months by the RSO or designee, unless specifically extended by the RSO.

Figure 3-1

The Cimarron Environmental Response Trust Organization



4.0 ALARA PROGRAM

4.1 Section Overview

This section describes the philosophy, requirements, and responsibilities of the Cimarron Site As Low As Reasonably Achievable (ALARA) program.

4.2 ALARA Policy

The Cimarron Site radiation protection program uses, to the extent practicable, procedures and engineering controls based upon sound radiation protection principles to achieve occupational doses and dose to members of the public that are ALARA. The Licensee is committed to providing resources such as personnel, training programs, engineering controls, monitoring devices, activity planning, administrative controls, etc. to achieve the goals of the ALARA principle.

Radiation Protection Procedure, RP-10, "ALARA Program," is the implementing procedure for the ALARA program. In addition, the Licensee encourages individuals working at the Site to provide input regarding improvements that would minimize dose and improve the safety and efficiency of activities.

4.3 ALARA Committee

At a minimum, the ALARA Committee meets once each calendar quarter.

4.3.1 ALARA Committee Responsibilities

The responsibilities of the ALARA Committee include:

- Ensuring that ALARA policy and regulatory compliance are integrated into all Site work activities as appropriate

- Reviewing and approving ALARA goals for the Cimarron Site (if individual monitoring is required)
- Reviewing the effectiveness of the ALARA Program (if individual monitoring is required)
- Reviewing plans for new activities to ensure that the ALARA principle has been considered
- Reviewing data from liquid effluent discharges to address the need to incorporate the ALARA principle
- Annually reviewing the RPP to ensure regulatory compliance and to incorporate any necessary changes
- Evaluating and approving changes to the D-Plan or the RPP in accordance with License Condition 27(e)

4.3.2 Annual ALARA Committee Report

The ALARA Committee ensures that a formal annual report is submitted to the NRC that includes:

- A description of all changes, tests, and experiments made or conducted pursuant to License Condition 27(e), including a summary of the safety and environmental evaluation of each action.
- Any D-Plan or RPP pages revised pursuant to License Condition 27(e).

A formal report shall also be submitted to the NRC annually if no changes, tests or experiments were approved by the ALARA Committee.

4.3.3 ALARA Committee Membership

License Condition 27(e) states:

"The ALARA Committee shall consist of a minimum of three individuals, one of whom shall be designated as the ALARA Committee chairman. Of these three designees, one shall have expertise in management and shall have managerial and financial responsibility for the decommissioning of the site; one shall have expertise in decommissioning and shall be responsible for site decommissioning, and one shall be the site Radiation Safety Officer or equivalent and shall ensure conformance to radiation safety and environmental requirements. The designee with managerial and financial responsibility shall be employed by the Licensee's Trustee. The designee for decommissioning of the site and the Radiation Safety Officer or equivalent, shall be retained by the Trustee. Except for the representative of management, ALARA Committee members may be consultants."

In accordance with this License Condition, the ALARA Committee shall consist of a minimum of three voting members:

- The Trust Administrator is a permanent (voting) member who has managerial and financial responsibility for the decommissioning of the Site.
- The Trustee PM is a permanent (voting) member who is responsible for Site decommissioning and groundwater remediation.
- The RSO is a permanent (voting) member of the ALARA Committee who ensures conformance to radiation safety and environmental requirements.
- The QAC is a non-voting standing member of the ALARA Committee who ensures conformance with quality assurance requirements.

The ALARA Committee is chaired by the RSO and reports directly to the Trust Administrator.

The Licensee is authorized to make certain changes to the NRC-approved D-Plan and RPP without NRC's approval, if these changes are consistent with the ALARA principle and the decommissioning process. The criteria for approval of these changes are stipulated in License Condition 27(e) and require ALARA Committee approval. Formal approval of such changes shall require a majority of the voting members and documented in minutes from the ALARA Committee meeting where these changes were approved.

Additional members may be nominated and approved by the three voting members identified in License Condition 27(e). These members shall be identified in radiation protection procedure RP-11, "ALARA Committee Reviews and Evaluations." Non-voting members may be included, as appropriate, to address technical issues such as quality assurance, decommissioning activities, health physics, hydrogeology, etc.

4.3.4 ALARA Committee Meetings

ALARA Committee meetings will include reports on the following aspects of decommissioning work:

4.3.4.1 ALARA Committee Responsibilities

4.3.4.2 Changes to the RPP or D-Plan, Experiments, Tests

4.3.4.3 Annual review of the RPP and ALARA Policy (typically conducted in the 4th quarter)

4.3.4.4 Annual report on D-Plan and RPP changes (1st quarter)

4.3.4.5 Radiation Protection Program

4.3.4.6 Radiological exposures/dosimetry/surveys

4.3.4.7 Compliance with license possession limits

4.3.4.8 ALARA goals and action levels

4.3.4.9 Radiation protection (RP) procedures/Desk instructions (RP-DI)

4.3.4.10 Instrumentation

4.3.4.11 Training and Qualification

4.3.4.12 Material Control & Accountability Program

4.3.4.13 Quality Assurance Program

4.3.4.14 Notices of deficiency

4.3.4.15 Quality Assurance Implementing Procedures (QAIP)

4.3.4.16 Audits and surveillances

4.3.4.17 Effluents

4.3.4.18 Compliance with OPDES Permit limits

4.3.4.19 Planned Work/Decommissioning Activities

4.3.4.20 Activity Plans (AP)

4.3.4.21 Sampling and Analysis Plan (SAP)

4.3.4.22 Chemical/Hazardous Material Concerns

4.3.4.23 Health and Safety Program

4.3.4.24 Radiological Waste Characterization and Disposal

5.0 ASSESSMENTS

5.1 Section Overview

Audits and/or surveillances provide a review of decommissioning and radiation protection activities to evaluate compliance with regulatory requirements, license conditions, the RPP, and radiation protection procedures. Audits and/or surveillances identify unsatisfactory performance and/or weaknesses in procedures, training, or work practices. The results of all audits and surveillances are reviewed by the ALARA Committee.

5.2 Audits

10 CFR 20.1101(c) requires that a licensee shall, at least annually, review the radiation protection program content and implementation. To satisfy this requirement, an annual audit is performed by the QAC and/or other individuals appointed by the Trustee PM. The audit is based upon various NRC guidance documents including Appendix H, NUREG-1556, Vol. 7, which provides sample audit forms to assist licensees in meeting this requirement. Audits shall be documented, as well as program changes resulting from audit findings or observations.

Corrective action for non-conformances and incidents is implemented through the "Notice of Deficiency" reporting process. Notices of Deficiency document failure to comply with specified requirements.

This process provides for the prompt identification of conditions adverse to quality, determination of their cause, and resolution of the specific conditions adverse to quality. A log of deficiencies and corrective actions is maintained to permit trending analysis if appropriate. The trend analysis can be used to identify timely corrective actions to prevent recurring problems

and improve performance. Deficiency reporting and the corrective action process are controlled by a single procedure under the QAPP.

5.3 Surveillances

Surveillances are observations of activities being performed. Surveillances of Site activities are done by, or under the direction of, the QAC and/or the RSO. The goal of surveillances is to determine whether or not an activity is being performed in accordance with applicable procedures, plans, accepted industry standards, etc. Surveillances of RPP-related activities shall be documented, as well as program changes resulting from findings or observations made during surveillances. Radiation protection surveillances are conducted once each calendar quarter at a minimum.

5.4 Records

Records of audits and surveillances are maintained in accordance with the QAPP.

Audit and surveillance records shall include the following information:

- The date(s) the audit/surveillance was conducted.
- Name of person(s) conducting the audit/surveillance.
- Audit/surveillance findings, corrective actions, and follow-up.

6.0 PERSONNEL MONITORING

6.1 Individual Monitoring of Occupational Dose

NRC regulation 10 CFR 20.1502 requires the licensee to monitor occupational exposures from both licensed and unlicensed radiation sources. Monitoring is required of any adult likely to receive, in 1 year from sources external to the body, a dose in excess of 10 percent of the Occupational Dose Limits for Adults and/or who are likely to receive, in 1 year, an intake in excess of 10 percent of the applicable annual limit on intake (ALI) in Table 1, Columns 1 and 2, of Appendix B to 10 CFR 20.1001-20.2402. Monitoring for minors is required when they are likely to receive, in 1 year, from radiation sources external to the body, a deep dose equivalent (DDE) in excess of 0.1 rem, a lens dose equivalent in excess of 0.15 rem, or a shallow dose equivalent to the skin or the extremities in excess of 0.5 rem and/or likely to receive, in 1 year, a committed effective dose equivalent (CEDE) in excess of 0.1 rem. Monitoring of declared pregnant women is required when they are likely to receive during the entire pregnancy, from radiation sources external to the body, a DDE in excess of 0.1 rem and/or likely to receive during the entire pregnancy, a CEDE in excess of 0.1 rem.

Personnel monitoring has not been performed at the Site since 2006 because there was no potential to receive a dose that would require monitoring under 10 CFR 20.1502. During the design of groundwater extraction and treatment systems, new work activities, such as groundwater processing, were evaluated to determine if they may result in exposure requiring personnel monitoring. The threshold dose for personnel monitoring will not be approached; accordingly, monitoring of workers is not required for external or internal occupational dose. Area radiation monitoring was established (Section 10.5) to confirm the results of this

evaluation. Air sampling during spent ion exchange resin handling activities will be performed as discussed in Section 6.6, below, and Section 10.8.

Two calculations were performed to determine the potential radiological conditions that may be encountered when the groundwater treatment system is operational. One calculation was performed to determine the potential intake from the groundwater processing operations. The other calculation was performed to determine potential external dose rates from spent resin vessels.

- Potential intakes from airborne exposure to uranium while handling spent resins are documented in Appendix A (EPM028-CALC-001, Potential Intake Calculation). Appendix A also provides the potential intake calculation for oral ingestion of uranium (Class Y), airborne exposure to Tc-99, and oral ingestion of Tc-99 (Class W). A sensitivity analysis to estimate potential intakes through inhalation of U-235 progeny (Th-231 and Pa-231) and U-238 progeny (Th-234 and Pa-234). These calculations demonstrate that the potential intakes of radioactive materials are very low. As discussed in section 10.8, worker intakes will be assessed based on potential exposure to Class Y uranium. The contamination control program described in this RPP is designed to ensure workers are not exposed to airborne radioactive material. The air sampling program described in the RPP is based on the results of these analyses and will be used to confirm conclusion of the calculations.
- Appendix B (EPM017-CALC-001, Dose Rate Near Uranium Treatment Train) provides the results of external dose rate calculations from spent resin vessels. Dose rates less than 0.3 mrem/hour were considered in the development of the radiation dose monitoring program described throughout this RPP.

These calculations were based on the 60% design of the groundwater treatment system. The potential intake calculation supported the decision that internal monitoring (e.g., bioassay) and respiratory programs were not needed at the Site. This calculation also informed the development of the air sampling program described in Section 10.7. The dose rate calculation supported the decision that personnel dosimetry was not required at the Site.

Both calculations will be reviewed at 90% design, updated, if necessary, and reevaluated to determine if the RPP should be updated. In addition, periodically through groundwater processing, these supporting calculations will be reviewed to ensure they reflect operational experience (including spent resin and effluent analyses) and determine if changes to the RPP are necessary. If additional activities are identified or planned, the radiological consequences of those activities will be evaluated to determine if personnel monitoring for occupational dose is required.

6.2 Occupational Dose Limits

NRC Regulation 10 CFR 20.1201 establishes a total effective dose equivalent (TEDE) limit and a total organ dose equivalent (TODE) limit for occupationally exposed adults. The TEDE is the sum of the DDE from external exposures and the CEDE from internal exposures. The TODE is the sum of the DDE and the committed dose equivalent (CDE) to the organ receiving the highest dose. The following annual dose limits apply to all Licensee employees, contractors, and visitors who receive occupational dose at the Cimarron Site.

Occupational dose is defined Section 16 of the RPP.

6.2.1 Occupational Dose Limits for Adults (10 CFR 20.1201)

- Whole Body - The more limiting of a TEDE equal to 5 rem or the sum of the DDE and CDE to any individual organ or tissue, other than the lens of the eye, equal to 50 rem.
- Skin of the whole body or skin of any extremity - A shallow dose equivalent equal to 50 rem.
- Lens of the Eye - A lens dose equivalent equal to 15 rem.

6.2.2 Occupational Dose Limits to Minors (10 CFR 20.1207)

- 10 percent of the corresponding limit for adults.

6.2.3 Occupational Dose Limits to Embryo/Fetus (10 CFR 20.1208)

- Dose to the embryo/fetus shall be limited to 500 mrem during the entire time of pregnancy of a declared pregnant woman. Substantial variations in dose rate over the gestation period shall be avoided.

6.3 Dose Limits for Individual Members of the Public (10 CFR 20.1301)

The TEDE received by individual members of the public from licensed operations shall not exceed 100 mrem in a year, exclusive of the dose contributions from background radiation, from any administration the individual has received, from exposure to individuals administered radioactive material and released under 10 CFR 35.75, or from voluntary participation in medical research programs. In addition, the dose in any Unrestricted Area from external sources shall not exceed 2 mrem above background in any one hour. Members of the public are not subject to the individual monitoring, record keeping, and reporting requirements of 10 CFR 20.

6.4 Determination of Prior Occupational Exposure

The occupational dose during the current year shall be determined and an attempt shall be made to obtain records of lifetime dose for all personnel who are likely to receive a dose in excess of 10% of the annual dose limit. The prior dose history shall be documented on Form NRC-4, or equivalent. Forms NRC-4 and NRC-5 and records used in their preparation shall be retained by the Licensee until the regulating agency terminates each pertinent license requiring this record and in accordance with the QAPP.

6.5 Personnel Monitoring for External Radiation

As discussed in Section 6.1, individual monitoring for external exposure is not expected to be required during groundwater extraction and processing and related activities. Passive area radiation monitoring using thermoluminescent dosimeters (TLDs) or optically stimulated luminescent dosimeters (OSLDs) will be performed to demonstrate that individuals will not exceed the requirements for individual monitoring provided in the RPP. However, individual monitoring devices will be assigned if any of the following conditions are encountered or expected to be encountered:

- Any individual likely to receive, from radiation sources external to the body, a dose in excess of 10 percent of the 10 CFR 20 occupational dose limits in a year.
- Any minor likely to receive, in 1 year, from radiation sources external to the body, a DDE in excess of 0.1 rem, a lens dose equivalent in excess of 0.15 rem, or a shallow dose equivalent to the skin or the extremities that exceeds 0.5 rem.
- Any declared pregnant woman likely to receive during the entire pregnancy, from radiation sources external to the body, a DDE that exceeds 0.1 rem.

When external exposure is determined by measurement with an external personal monitoring device, the DDE will be used in place of the effective dose equivalent, unless the effective dose equivalent is determined by a dosimetry method approved by the NRC. Dosimetry devices shall be processed by a laboratory or vendor maintaining accreditation by the National Voluntary Laboratory Accreditation Program (NVLAP).

If the need to perform external monitoring for workers is identified, RP procedures will be implemented that consider guidance provided in Regulatory Guides 8.4, Rev. 1, 8.28, Rev. 0, and 8.34, Rev.0, as applicable. The following information will be addressed in these procedures:

- A description of the individual-monitoring devices that will be provided to workers who meet the criteria in 10 CFR 20.1502(a) and 20.1601 for external exposures.
- The type, range, sensitivity, and accuracy of each individual-monitoring device.
- Use of extremity and whole body monitors when the external radiation field is non-uniform.
- When audible-alarm dosimeters and pocket dosimeters will be provided, and a description of their performance specifications.
- How external dose from airborne radioactive material is determined.
- The procedure to ensure that surveys necessary to supplement personnel monitoring are performed.
- Action levels for workers' external exposure, including the technical bases and actions to be taken when they are exceeded.

6.6 Internal Exposure Monitoring

Based on anticipated radiological work involving extraction and treatment of groundwater at the Site, internal exposure monitoring is not warranted. If radiological conditions change or

evaluation of the final groundwater processing equipment design indicates that an individual worker could be exposed to 2% of the ALI for Class Y uranium in a year, then bioassay shall be performed. Bioassay shall be performed whenever a calculated intake of 40 Derived Air Concentration (DAC)-hours could have occurred in any one incident based on air sampling data, accident conditions, equipment failure, external contamination, or other conditions. Bioassay sampling shall also be performed whenever it is likely that an individual may have received an intake of 10 milligrams of uranium in any one week. Bioassay shall be considered upon termination of all Radiation Workers who may have had intakes of radioactive materials. The need for bioassay sampling shall be determined by the RSO or designee. Requirements for the determination of internal exposure are provided in 10 CFR 20.1204.

If the need for internal monitoring is identified, RP procedures will be implemented that consider guidance provided in Regulatory Guide 8.9, Rev. 1, Regulatory Guide 8.15, Rev. 1, Regulatory Guide 8.34 Rev. 0, and Regulatory Guide 8.36, Rev. 0, as applicable. The following information will be addressed in these procedures:

- How worker intakes are determined using measurements of quantities of radionuclides excreted from or retained in the human body. Specifically, the procedures will address how frequencies for bioassay measurements for baseline, periodic, special, and termination assays are assigned.
- How radioactivity measured in the human body by bioassay techniques are converted into worker intake; and action levels for bioassay samples, actions to be taken when they are exceeded, and their technical bases.
- How worker intakes are determined by measurements of the concentrations of airborne radioactive materials in the workplace. Specifically, the procedures will address how airborne concentrations of radioactivity are measured; how airborne concentrations are

converted to determine intakes; action levels for a worker's intake based on dose, and actions to be taken when they are exceeded; and action levels for a worker's intake based on chemical toxicity if soluble uranium is present in the work area.

- How worker intakes, for an adult, a minor, and a declared-pregnant woman are determined using any combination of the measurements above.
- How worker intakes are converted into committed effective dose equivalent (and organ-specific committed dose equivalent), Including how intake of radioactivity by a DPW will be converted into dose to the embryo/fetus.

6.7 Declared Pregnant Woman Exposure Policy

Based on recommendations of the NCRP and on regulatory requirements, controls are established for the protection of the embryo/fetus during a declared female workers pregnancy. These controls shall ensure compliance with regulatory requirements and protect the rights of the female worker.

Declaration of pregnancy is at the discretion of the woman (medical proof is not required). Any woman who does not declare her pregnancy shall be subject to the normal occupational dose limits and shall not be subject to special controls or treatment with respect to work assignments involving exposure to radiation even if she is pregnant. The Licensee shall ensure the dose to the embryo/fetus of a declared pregnant woman does not exceed regulatory limits due to occupational dose during the entire pregnancy.

If internal monitoring for declared pregnant workers is determined to be necessary, procedures for determining dose to the embryo/fetus will be developed and implemented. Dose to the embryo/fetus will be determined based on guidance provided in Regulatory Guide 8.36 and ICRP Publication 88.

6.8 Summation of Internal and External Dose

Internal and external doses are summed whenever positive doses are measured. Procedures will be used to document the methodology for the summation of internal and external doses to workers and internal dose contribution from maternal intakes to the embryo/fetus of a declared pregnant worker. If internal and external monitoring is performed, RP procedures will be implemented that consider guidance provided in Regulatory Guides 8.7, Rev. 4, 8.34, Rev. 0, and 8.36, Rev. 0, as applicable. The following information will be addressed in these procedures:

- How the internal and external monitoring results are used to calculate TODE and TEDE doses to occupational workers.
- How internal doses to the embryo/fetus, which is based on the intake of an occupationally exposed, declared-pregnant woman, will be determined.
- A description of the monitoring of the intake of a declared-pregnant woman if determined to be necessary.
- A description of the program for the preparation, retention and reporting of records for occupational radiation exposures.

6.9 ALARA Dose Goals

As discussed in Section 4.3, ALARA dose goals will be set if individual monitoring is required. Until such time, the annual Administrative Dose Goals for the Site is effectively 100 mrem TEDE. If the Administrative Dose Goals are exceeded without prior authorization, the RSO or designee shall investigate to determine the cause and prepare a written report to document the results of the investigation and any corrective actions taken or planned.

6.10 Personnel Exposure Reports

An annual report of the individual radiation dose received shall be sent to each worker who was issued individual dosimetry and/or was subject to the requirements for monitoring as specified in Section 6.1. When requested by an individual, a written exposure report shall be provided to each such individual within 30 days of the request or within 30 days of exposure determination, whichever is later.

Internal and external doses shall be summed whenever positive doses are measured. The dose to the lens of the eye, skin, and extremities are not included in the summation. Intakes through wounds or skin absorption shall be evaluated and, to the extent practical, accounted for in summation of internal and external doses independent of intakes by ingestion or inhalation.

6.11 Dosimetry Records

Records of individual monitoring shall be kept in accordance with 10 CFR 20.2106 and the QAPP. These records shall be updated at least annually for any radiation monitoring data collected. All radiation exposure records shall use the units curie, rem, rad, or multiples thereof.

Records of doses received by individuals for whom monitoring was required pursuant to 10 CFR 20.1502, and records received during planned special exposures, accidents, and emergency conditions shall be maintained. These records must include the following, when applicable"

- The DDE to the whole body, lens dose equivalent, shallow-dose equivalent to the skin, and shallow dose-equivalent to the extremities;
- The estimated intake of radionuclides (10 CFR 20.1202);
- The CEDE assigned to the intake of radionuclides;

- The specific information used to assess the CEDE pursuant to 10 CFR 20.1204(a) and (c), when required by 10 CFR 1502;
- The TEDE when required by 10 CFR 20.1202; and
- The total of the DDE and the committed dose to the organ receiving that highest total dose.

Procedure RP-19, "Dosimetry Records," provides for the preparation, retention, and reporting of records of occupational dose. This procedure addresses: recordkeeping frequency (10 CFR 20.2106(b)), recordkeeping format (10 CFR 20.2106(c)), privacy protection (10 CFR 20.2106(d)). RP-19 provides instructions for maintaining records of embryo/fetus with those of the mother, including the declaration of pregnancy (10 CFR 20.2106(f)).

7.0 RADIATION PROTECTION INSTRUMENTATION

7.1 Calibration

Calibration of radiation monitoring, counting, and air sampling instruments shall be performed in accordance with the manufacturers' recommendation unless otherwise approved by the RSO. These calibrations shall be consistent with regulatory requirements.

The calibration frequency for portable radiation monitoring instruments and portable air sampling equipment shall be at least every 12 months. Benchtop smear/sample instrumentation shall be calibrated at least annually.

Calibration of radiation protection instruments is performed by an approved vendor except for air samplers. Air sampler flow rate indicators are calibrated in accordance with manufacturer's recommendation using a reference air flow calibrator (calibrated annually by an approved vendor). The air sampler flow rate indicator is adjusted as necessary to ensure reported values are correctly stated and valid under the actual operating conditions of the air sampler. Detailed instructions for air sampler calibrations are provided in procedures.

Instruments used to perform release surveys must be calibrated using National Institute of Science and Technology (NIST) traceable, or equivalent, standards for energies and geometries similar to material being released. The energy dependence of the instruments to alpha, beta, and gamma radiation, as applicable, shall be known and documented.

7.2 Operation and Response Tests

Operation and response tests of radiation monitoring, counting, and air sampling instruments, shall only be performed by personnel trained in the use of the instrument and following approved procedures. At a minimum, on a daily basis, prior to use, each radiation protection instrument shall be subject to the following:

- The instrument and detector are in good physical condition.
- Verification of current calibration.
- Checking the battery, if applicable, and replacing the battery, if necessary.
- Source check.
- Background determination.

Detailed instructions for each radiation protection instrument on operation and response tests are provided in radiation protection procedures.

7.3 Maintenance and Repair

Individuals authorized by the RSO may perform routine maintenance and limited field repairs, such as replace batteries, cables, or mylar windows. Other maintenance and repair of radiation protection instrumentation shall be performed by an approved vendor. All maintenance and repair affecting calibration shall be documented.

7.4 Quality Control/Quality Assurance

Quality Control (QC) measures for instruments shall be established and implemented to maintain reliability of counting results and sensitivities.

7.5 Radiation Protection Instrumentation Inventory

Table 7-1 provides a list of equipment available to perform radiological surveys at the Site. Minimum quantities that need to be available when groundwater processing commences are provided for each instrument. The specified quantity may not be present when an instrument is sent for calibration or repair. The RSO or designee will determine if there is a need to rent or purchase additional instruments when an instrument is being calibrated or repaired.

Radiation protection instrumentation and exempt quantity check sources will be stored in a secure storage area at the Western Area Treatment Facility.

7.5.1 Instrument Procedures

Procedures provide instructions for using specific instruments, including the following information:

7.5.1.1 Instrumentation storage, calibration, and maintenance for instruments used in field surveys, including analyses of smears and air samples collected during surveys. Maintenance activities performed at the Site include battery replacement, mylar window repair, and cable replacement, which are performed in the HP Office in the WATF.

7.5.1.2 MDC or MDA (at the 95 percent confidence level) for each type of radiation to be detected, as appropriate

7.5.1.3 Determination of MDC for scanning surveys (if used at the SITE

7.5.1.4 The prescriptive use of borosilicate glass fiber media for air sampling and the requirement to perform an analysis if different filter media is

used., including the need to adjust the equation for determining the MDC for air sampling as discussed in the Note in Section 7.5.1.

7.5.1.5 Determination of site-specific counting efficiency for contamination survey equipment.

7.5.1.6 Quality control for instruments consistent with the manufacturer's instructions and be consistent with regulatory requirements.

7.5.1.7 Calibration of portable survey equipment by a qualified vendor in accordance with ANSI N323A-1997, *Radiation Protection Instrumentation Test and Calibration, Portable Survey Instruments*. Later versions of this standard are also acceptable.

7.5.2 Contamination Survey Instrument Counting Efficiency

Counting efficiency for instruments used to measure total surface contamination or analyzing smears for removable contamination is determined consistent with manufacturers' recommendations. Typically, a site-specific counting efficiency is determined using sources identified in Table 7-2. The manufacturer participates in a National Institute of Science and Technology (NIST) measurement assurance program to establish and maintain an implicit traceability for a number of nuclides, based on the blind assay (and later NIST certification) of Standard Reference Materials (as in NRC Regulatory Guide 4.15).

a. For alpha counting efficiency, the efficiency is based on a 10-minute count of the Th-230 source (see Table 7-2). The net counts (gross counts minus background counts) is used to determine the net counts per minute. The net

counts per minute are divided by the 2π surface emission rate (α per minute) to determine the alpha counting efficiency.

b. For beta counting efficiency, the efficiency is based on a 10-minute count of the Tc-99 source (see Table 7-2). The net counts (gross counts minus background counts) is used to determine the net counts per minute. The net counts per minute are divided by the 2π surface emission rate (β per minute to determine the beta counting efficiency.

c. The RSO may provide alternative methods for determining counting efficiency, which could include use of the counting efficiency provided on the instrument calibration certificate. Other alternative methods may be evaluated for use at the Site. If selected, RSO approval of the alternative method shall be documented with the instrument records.

NOTE:

1. The minimum detectable activity (MDA) for portable survey instruments is calculated by the following equation:

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

Where:

R_b is background count rate (counts/minute)
T_s is sample count time (minutes)
T_b is background count time (minutes)
E is instrument efficiency (counts/disintegration)
P_A is the probe area(cm^2)

This equation is equivalent to Eq 3-11 of NUREG-1507. The surface efficiency is taken into account in the determination of the instrument efficiency. The following surface efficiencies factors are used in the development of the instrument efficiency:

- Alpha emitters – 0.25
- Beta emitters – 0.3
- Gamma emitters – 1.0

A surface efficiency factor is not applied to measurements of wipe sample or air samples.

1. For air sampling, the equation above is adjusted to account for the volume of air sampled. The minimum detectable concentration (MDC) for air sampling is calculated by the following equation:

$$MDC\left(\frac{\mu Ci}{mL}\right) = \frac{\frac{3}{T_s} + 3.29\sqrt{\frac{R_b}{T_s} + \frac{R_b}{T_b}}}{EVC}$$

Where:

R_b is background count rate (counts/minute)
T_s is sample count time (minutes)
T_b is background count time (minutes)
E is instrument efficiency (counts/disintegration)
V is volume of air of air sampled (mL)
C is conversion of µCi to dpm (i.e., 2.22E+06)

For the Ludlum 3030E, the typical minimum detectable activity, for a one-minute count, is 16 dpm for alpha (Th-230) and 60 dpm for beta (Tc-99) or 7.2E-06 µCi and 2.7E-05, respectively. In the case of a four-hour air sample taken at 56 Lpm, the alpha MDC is 5.4E-13 µCi/mL and beta MDC is 2.0E-13 µCi/mL.

The filter media for air sampling is borosilicate glass fiber, unless otherwise specified by the RSO. As the collection efficiency for borosilicate fiber is >95%, no correction for filter efficiency is required as discussed in Section 6.2 of NUREG-1400. If other filter media with a collection efficiency is used, a correction factor will be used to calculate MDC, as discussed in Section 6.2 of NUREG-1400.

2. Beta Scan MDC – To determine the minimum detection concentration (MDC) in dpm/100cm^2, for beta scanning surveys, the following equation (NUREG-1507 Section 6.2.4) is used:

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

Where:

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

As an example, calculation, the Scan MDC using the Ludlum 2360 with a Model 43-93 detector is determined considering a background level typically encountered at the Site averaging about 165 cpm and 95% correct detections and 60% false positive rates resulting in a d' of 1.38. The scan rate A surveyor moving a detector at one detector width per second across the center of a hypothetical 100 cm^2 hotspot will have an observation interval of about 1 second (per NUREG-1507).

$$b_i = 165\ cpm \times 1\ sec \times \left(\frac{1\ min}{60\ sec}\right) = 2.75\ counts$$

$$Scan\ MDC = \frac{1.38 \times \sqrt{2.75} \times \left(\frac{60\ sec}{\frac{1\ min}{1 sec}}\right)}{\sqrt{0.5} \times 0.115 \times 0.5 \times \left(\frac{100\ cm^2}{100\ cm^2}\right)} = 3378 \frac{dpm}{100\ cm^2}$$

3. Alpha Scan MDC – To determine the minimum detection concentration (MDC) in dpm/100cm^2, for alpha scanning surveys, NUREG-1575, Rev. 2, provides Equation 6-14 for determining the scan MDC for

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

Where,

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

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

ε_t is the detector efficiency (4π)

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

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

Although examples of scan MDCs have been provided in the previous note, scan MDCs are not relevant to unrestricted release surveys at the Site. License Condition 27.c provides specific unrestricted use criteria that must be used for soils and soil-like material. The License Condition provides specific values for surfaces of buildings and equipment. Compliance with these criteria is based on static measurements and not the results of a scanning survey. Material that exceeds the averaging criteria must be removed and shipped off-site to a licensed low-level radioactive waste disposal site.

Review and evaluation of instrumentation operability shall be performed on an on-going basis by the RSO or designee.

Table 7-1

Radiation Protection Instrument List

Man.	Model	Probe	Min. Quant.	Description
Ludlum	12	44-9	2	Instrument Type: Handheld analog ratemeter with a GM pancake-type detector. Use: Contamination surveys, equipment and materials restricted release, personnel frisking. Ranges: 0-500 cpm; 0-5,000 cpm; 0-50,000 cpm; 0-500,000 cpm Counting Modes: Ratemeter. Alarm Set Point: N/A
Ludlum	19	N/A	2	Instrument Type: Gamma micro-R meter (0 to 5000 µR/hr). Use: Low-level gamma dose rate measurements. Ranges: 0-25 µR/hr; 0-50 µR/hr; 0-250 µR/hr; 0-500 µR/hr; 0-5,000 µR/hr; Counting Modes: Ratemeter. Alarm Set Point: N/A
Ludlum	2360	43-93	3	Instrument Type: Alpha-Beta Ratemeter, Scaler, and Data Logger with a dual phosphor scintillation probe. Use: Contamination surveys, material and equipment unrestricted and restricted release, personnel frisking. Ranges: 0-500 cpm; 0-5,000 cpm; 0-50,000 cpm; 0-500,000 cpm Counting Modes: Ratemeter, scaler, data logger. Alarm Set Point: N/A
Ludlum	3030E	43-10-1	1	Instrument Type: Dual channel, scaler-type sample counter with a dual phosphor detector. Use: Alpha and beta smear counting, air sample analysis. Ranges: Counting Modes: Count times are specified by procedure. Alarm Set Point: N/A
Ludlum	2221	44-10	2	Instrument Type: Handheld ratemeter and scaler with an analog display for viewing the count rate with a 2" X 2" NaI(Tl) scintillator. Use: Walk-over (qualitative) surveys. Ranges: 0-999,999 counts Counting Modes: cpm and dpm, data logging Alarm Set Point: Not used.

Table 7-1 - continued

Radiation Protection Instrument List

Air Sampling Equipment				
Man.	**Model**	**Filter Head**	**Min. Quant.**	**Description**
RADEco	AVS-28A	2500-42	2	Portable, low volume, continuous flow air sampler with a 47 mm diameter open face filter head Air Flow Rate: 15-100 Lpm Elapsed Time Meter: 99,999 hours and 59 minutes

Table 7-2

Cimarron Site Radioactive Check Sources

Manufacturer	**Source No.**	**Radio-nuclide**	**Source Activity**		**Surface Emission Rate**	**Dimensions**
			nCi	**Bq**		
Certified Sources (used for determining instrument counting efficiency and performing response checks)						
Eckert & Ziegler	T8-416	Th-230	4.875	180.4	5465 α/min	Disk (47 mm outside diameter X 0.76 mm thick)
Eckert & Ziegler	T8-417	Tc-99	4.965	183.7	6062 β/min	Disk (47 mm outside diameter X 0.76 mm thick)
Check Source (may be used only for performing instrument response checks)						
N/A	118	Cs-137	1000	N/A	N/A	N/A

8.0 ACCESS CONTROL

8.1 Section Overview

This section provides the access control requirements for entry into and exit from RAs. Access control is designed to ensure that individuals have appropriate qualifications, training, and authorization for entry. Access control requirements are applicable to personnel, contractors and visitors who enter RAs. RAs are areas within the Site boundary for which access is controlled for the purpose of protecting individuals against undue risk from exposure to radiation and/or radioactive materials.

The tentative designation of RAs during initial groundwater treatment for the Western Area Treatment Facility is provided in Figure 8-1. The Burial Area #1 (BA#1) influent could contain greater than ten times the 10 CFR 20 Appendix C quantity for U-235 and will be designated as a Restricted Area until quantities of license material in the tank can be demonstrated through sampling and calculation of total activity to be less than ten times the 10 CFR 20 Appendix C.

NOTE: This figure is an annotated version of a drawing taken from the D-Plan. RAs will be periodically reviewed and may be expanded, reduced, or reconfigured based on RSO evaluation of potential exposure to radioactive material. Additional areas may be designated as RAs if appropriate. The RPP will be updated accordingly.

RAs will be established based on the potential for accumulating radioactive material greater than ten times the 10 CFR 20 Appendix C quantities or requiring posting as Radiation Areas, High Radiation Area, Contaminated Area, or Airborne Radioactivity Areas. No High Radiation Areas are anticipated based on the groundwater treatment facility design. Documentation of RAs and any changes to designated RAs shall be maintained as a decommissioning record.

8.2 Restricted Area Access Controls

Access to any RA shall be permitted only to personnel trained or escorted as discussed in Section 2. Personnel who enter RAs may be required to wear dosimetry. RAs include Radioactive Material Areas, Radiation Areas, High Radiation Areas, Contaminated Areas, and Airborne Radioactivity Areas. RAs can be controlled through the use of guards, barriers, fences, signs, gates, or doors.

RA boundaries shall be defined by the use of postings, barriers, walls, tape, ropes, markings, or locked doors. A log of personnel entry and exit to any RA, other than Radioactive Material Areas, at the Site will be maintained by the RSO or designee. A log of personnel entry into areas posted solely as Radioactive Materials Areas is not required.

8.2.1 Contaminated Area Access and Control

In addition to the controls discussed above, access to Contaminated Areas shall include an identified entry/exit point where a step-off pad will be placed immediately outside the boundary of the Contaminated Area. Radiation protection procedures shall include the following information regarding contamination area controls:

- A description of the procedures to control both access to and stay time in Contaminated Areas by workers if they are needed.
- A description of surveys to supplement personnel monitoring for workers during routine operations, maintenance, clean-up activities, and special operations.
- A description that describes contamination action limits (i.e., actions taken either to decontaminate a person, place, or area, or to restrict access, or to modify the type or frequency of radiological monitoring). These action limits are provided in Table 12-1.

- A description of radiological contamination guidelines (Table 12-1) for specifying and modifying the frequency for each type of survey used to assess the reduction of total contamination.

8.3 Posting and Labeling Requirements

Posting of areas within each RA shall be performed in accordance with 10 CFR 20 Subpart J. Containers of radioactive materials shall be labeled in accordance with 10 CFR 20.1904. Exceptions to posting requirements found in 10 CFR 20.1903 and exceptions to labeling requirements found in 10 CFR 20.1905 shall be approved by the RSO or designee.

Signs used for posting radiological areas within an RA shall include the wording provided in Table 8-1 when the associated requirements are expected to be encountered or expected to be encountered:

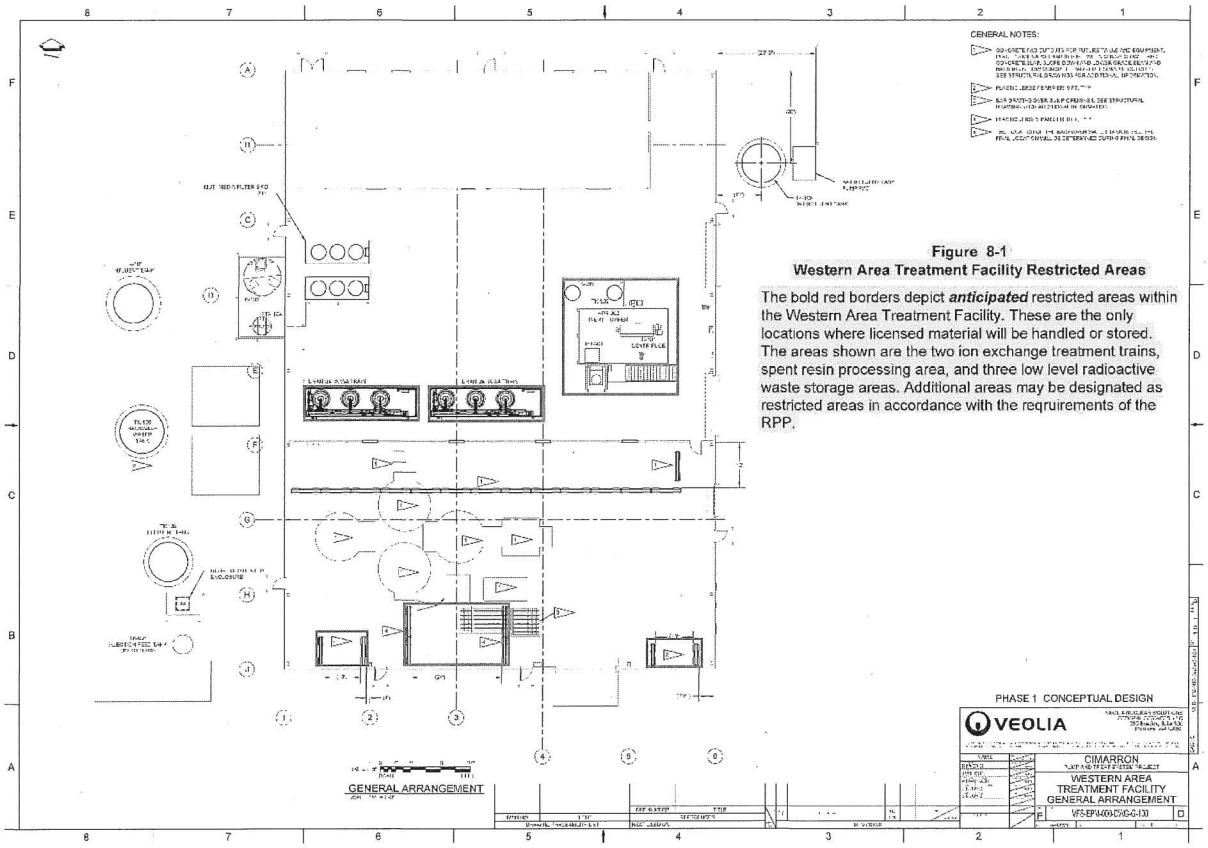
Table 8-1

Radiological Posting Requirements

POSTING WORDING	REQUIREMENT
"CAUTION, RADIATION AREA"	Accessible area in which radiation levels could result in an individual receiving 5 mrem in one hour 30 cm from the radiation source or surface that the radiation penetrates.
"CAUTION, HIGH RADIATION AREA" or "DANGER, HIGH RADIATION AREA"	Accessible area in which radiation levels could result in an individual receiving 100 mrem in one hour 30 cm from the radiation source or surface that the radiation penetrates.
"CAUTION, AIRBORNE RADIOACTIVITY AREA" or "DANGER, AIRBORNE RADIOACTIVE MATERIALS AREA"	Licensed airborne radioactive materials in a room, enclosure, or area exists in concentrations exceeding the DACs specified in 10 CFR 20 Appendix B, Table I, or when an individual present in the area without respiratory protective equipment could exceed, during the hours an individual is present in a week, an intake of 0.6% of the ALI or 12 DAC–hours.
"CAUTION, CONTAMINATED AREA"	Accessible area in which removable contamination levels exceed 1,000 dpm/100 cm^2 beta/gamma contamination or 1000 dpm/ 100 cm^2 alpha contamination.
"CAUTION, RADIOACTIVE MATERIAL(S)" or "DANGER, RADIOACTIVE MATERIAL(S)"	Areas or rooms in which there is used or stored an amount of licensed radioactive material exceeding 10 times the quantity of such material in 10 CFR 20 Appendix C.

Figure 8-1

Western Area Treatment Facility



9.0 RADIOLOGICAL WORK CONTROLS

9.1 Section Overview

Radiological work within RAs is controlled through two mechanisms: Site procedures and activity plans.

- Site procedures include quality assurance procedures, radiation protection procedures, sampling and analysis procedures, standard operating procedures, waste management procedures, etc. Site procedures cover routine work or repetitive tasks that may include radiological work. Any necessary radiological controls are included in Site procedures.
- Activity plans cover non-routine work activities and include information on the conditions that exist in the work area and radiological and non-radiological safety requirements. To ensure compliance with the RPP and regulatory requirements, activity plans involving radiological work must include the information identified in Section 9.2.

Work within posted High Radiation Areas, Airborne Radioactivity Areas, and Contaminated Areas, or requiring the use of respiratory protection or protective (i.e., anti-contamination) clothing shall be controlled through the use of an activity plan unless specifically authorized by the RSO or designee.

9.2 Activity Plan Requirements

The activity plan job description and job location shall be consistent with the activities or task to be performed. The activity plan shall identify potential radiological hazards, methods to address radiological hazards, and protective equipment needed for the work. Activity plans shall, as a minimum, include:

- A description of the work,

- Anticipated radiological conditions,
- Reference to applicable radiation protection procedures,
- Radiation safety requirements,
- Required personal protective clothing and equipment,
- Radiological survey and/or monitoring requirements,
- Radiation safety training requirements,
- Special radiation protection sampling requirements.

9.2.1 Activity Plan Approval/Closeout

Activity plan approval and closeout is addressed in the QAPP and implementing procedures.

9.2.2 Activity Plan Training

Training and qualifications for individuals working under an activity plan are addressed in the QAPP. All Radiation Workers operating under an activity plan are required to review and comply with the activity plan.

9.2.3 Record Keeping

Activity plan records are maintained in accordance with the QAPP.

9.3 Receipt of Potentially Contaminated Tools, Equipment, Parts, and Material

Tools, equipment, parts, and material that have been used at oil and gas facilities may be contaminated with naturally occurring radioactive material (NORM) or other radioactive material used as tracers. Qualified individuals shall perform receipt surveys to document the

radiological conditions of all tools, equipment, parts and equipment potentially used at oil and pipeline facilities or sites prior to use at the Cimarron Site.

Procurement specifications for tools, equipment, parts, and material previously used at oil and gas facilities shall require cleaning of these items prior to shipment to the Site.

The Site cannot receive tools, equipment, parts, and material that are potentially contaminated with radioisotopes other than NORM or uranium.

Receipt surveys will also be performed on sample shuttles/coolers provided by vendor laboratories used to ship samples for analysis from the Site. Each shuttle/cooler will be surveyed for total contamination and removable contamination.

If the receipt survey detects fixed or removable contamination or if dose rates two times background are detected, these items shall be segregated and the RSO and Trustee PM will be notified. The RSO and Trustee PM will determine disposition of these items.

10.0 RADIATION PROTECTION SURVEYS

10.1 General Requirements

Radiological survey information is used to:

- assist in the development of activity plans,
- inform individuals of the radiological conditions/hazards in the area,
- evaluate the need for area postings,
- identify needed personnel protective equipment,
- verify the effectiveness of engineering and administrative controls,
- ensure personnel exposures to radiation and radioactive materials are maintained ALARA,
- determine the decommissioning status of material, equipment, and/or environmental media, and
- determine compliance with regulatory and/or license criteria.

Radiation and contamination surveys, air sampling, and sample collection will be performed as appropriate to assess radiological conditions and to establish specific radiological controls for work to be performed. Radiation protection surveys shall be conducted in accordance with approved procedures or desk instructions.

Two types of dose rates measurements are typically used during routine and job coverage surveys:

- Contact dose rates are used to locate and identify radiation levels detected within 1 cm (0.5 in) from the surface being surveyed.

- General area dose rates are used to identify radiation levels detected at approximately 30 cm (1 ft) from the surface being surveyed.

Certain exposure rate measurements must be taken at one meter (3.3 ft) from the surface of material or equipment to ensure the criteria for unrestricted use are satisfied prior to being released from license control. Dose rate and exposure rate measurement requirements are addressed in approved procedures or desk instructions.

Surveys for removable and direct contamination are performed to detect and/or quantify radioactive contaminants. Removable contamination surveys are performed when appropriate to ensure that radioactive contamination has not inadvertently spread. For direct contamination measurements on surfaces and individuals, typically, the probe is moved at a speed of about one probe width per second over the surfaced being measured for contamination at a distance of 0.5 (for beta/gamma) inches/0.25 inches (for alpha) or less from the surface. This method allows for identification of increased count rate where a static measurement is taken (30 second count time). Further instructions are provided in approved procedures.

NRC Regulatory Guide 8.25, "Air Sampling in the Workplace" provides an acceptable method for meeting certain survey and dose assessment requirements of 10 CFR 20. Air samples shall be collected whenever the airborne radioactivity levels are expected to exceed 10 percent of the DAC as listed in Appendix B, Table 1, "Occupational Values," of 10 CFR 20.

Breathing zone air sampling shall be performed whenever respiratory protection devices are worn by personnel for radiation protection purposes. If air sample data indicates a measured level greater than 40 DAC-hours in any shift or operation, whichever is shorter in time duration, the RSO or designee shall conduct an investigation and identify corrective actions to that are

needed to reduce airborne radioactivity levels. The RSO or designee shall work with the Activity Leader to implement the necessary corrective actions for reducing airborne radioactivity levels.

Air sample collection media shall be appropriate to address the radionuclide mixture(s) present. The analysis of air samples (including preliminary field screening) shall be performed in a timely and expeditious manner.

10.2 Routine Surveys

Routine radiological surveys are performed at a frequency that is consistent with the existing and potential hazards and activities planned in the RAs. The following radiation dose rate and contamination survey frequencies ensure area hazards are adequately characterized:

1. Weekly, in office space located in areas surrounding or adjacent to RAs, other than Radioactive Material Areas, where the potential exists for external radiation exposure or contamination spread.
2. Weekly, in routinely occupied RAs, other than Radioactive Material Areas.
3. Monthly, or upon entry if entries are less frequent than monthly, for Radioactive Material Areas.

During routine contamination surveys, if contamination levels exceed action levels discussed in Table 12-1, the RSO or designee will determine the cause for the contamination and determine appropriate corrective actions, including decontamination, increasing the frequency of surveys, need for additional engineering or administrative controls, etc.

10.3 Job Coverage Surveys

Job-coverage surveys are specified in activity plans and routine operations procedures. The types of radiological surveys (i.e., radiation, surface contamination, airborne radioactivity), frequency (e.g., number of times during a shift, at a specific step in the activity, etc.), and location are determined by the RSO or designee based on the radiological hazards associated with the work to be performed. Special survey requirements may be provided by the RSO or designee, when needed.

10.4 Investigative Surveys

Investigative surveys shall be performed as soon as practicable following the discovery or indication of abnormal radiological conditions.

10.5 Final Status Surveys

Final status surveys will be required to support license termination. During the post-remediation monitoring period, a final status survey plan will be developed and submitted for approval by the NRC.

10.6 Personnel Contamination Monitoring

Personnel contamination monitoring (frisking) shall be routinely performed prior to exiting RAs that have the potential for spreading contamination or per activity plan or procedural requirement. At a minimum, hands and feet shall be frisked when exiting these areas. Documentation of routine personnel frisking is maintained in field notes maintained by the HP Technician. Notification of the RSO or designee is required when personnel contamination in excess of twice the background count rate is detected. RP procedures provide specific

instructions approved by the RSO for performing, documenting, and reporting personnel contamination monitoring reports.

10.7 Area Radiation Monitoring

The RSO or designee will determine when and where area radiation monitoring is appropriate. Area radiation monitoring may be performed using either passive devices, such as dosimeters (e.g., thermoluminescent or optically stimulated luminescent) or real-time radiation monitors. Dosimeters are posted at the Cimarron Site to confirm that no occupational worker is likely to receive 100 mrem DDE in a year.

10.8 Air Monitoring

Air monitoring is required whenever airborne radioactivity levels are expected to exceed 10 percent of the DAC for Class Y uranium as listed in Appendix B, Table 1, "Occupational Values" of 10 CFR 20. Considering the types of work activities described in the D-Plan, airborne suspension of licensed radioactive material is not anticipated to generate airborne radioactivity approaching 10% of a DAC. However, the D-Plan requires that General Area (GA) air sampling, using either low or high-volume portable air samplers, will be performed throughout the resin unloading and packaging process for at least the first three resin exchanges. Lapel samplers will be used in conjunction with the GA samplers to demonstrate that GA samplers are representative of the air breathed by workers. Following analysis of the air sampling results from each of these resin exchanges, the RSO will determine the need for and frequency of additional air sampling and types of air sampling to be performed (e.g., GA or lapel).

NOTE: A prospective evaluation of potential intake during groundwater processing operations was performed. The calculation supporting this evaluation is provided in Appendix A of the RPP. This calculation was based on the 60% design of the groundwater treatment system and supported the decision that internal monitoring (e.g., bioassay) and respiratory programs would not be needed at the Site. The evaluation also informed the development of the air sampling program described in Section 10.8. The supporting calculation will be reviewed at 90% design, updated, if necessary, and re-evaluated to determine if the RPP should be updated. In addition, periodically through groundwater processing, the supporting calculation will be reviewed to ensure it reflects operational experience and to determine if changes to the RPP are necessary.

The results of this evaluation indicate that continuous air monitors are not needed as the potential for an individual to be exposed to 40 DAC-hours in week does not exist at the Cimarron Site. Updates to this calculation are reviewed to ensure this conclusion remains applicable.

10.8.1 Selection and Placement of Air Samplers

Selection of air samplers is based on the following criteria:

10.8.1.1 GA air sampling will be accomplished by using portable air samplers, as discussed, above. Sampling heads will be placed within the breathing zone to ensure that the air sample is representative of the air breathed by the individual worker.

10.8.1.2 GA air samplers typically sample at a rate of approximately 3-25 liters per minute (lpm) (less than 1 cubic foot per minute (cfm)) for a low-volume sampler to 1900 lpm (70 cfm) for a high-volume sampler. Based on the nature of the low enriched uranium encountered, the detection capability

of the air sampling equipment and associated radiological analysis (e.g., sample counting) will be used to determine the total volume of air needed to be collected to ensure that 10% of the DAC can be detected. The enrichment of the uranium will be based on either the actual enrichment being collected on the resin or a conservative basis (i.e., 5%). This calculation will be documented in a Site procedure or technical basis document. As the actual enrichment of recovered uranium, the 10% DAC value may be recalculated minimum collection times will be determined so adequate sensitivities are achieved for a given monitoring period.

10.8.1.3 The need for air sampling will be prospectively determined based on the final process system design and potential for generation of airborne radioactivity. Due to the chemical and physical nature of the uranium-bearing media (e.g., water and moist ion exchange resin), minimal, if any airborne radioactivity is expected to be generated. Engineering and physical controls incorporated into the process equipment design will also be considered in determining the need for air monitoring.

10.8.1.4 The frequency of calibration of the flow meters on the air samplers will be based on manufacturers' recommendations (typically annually).

10.8.1.5 Specific action levels (i.e., specific projected or actual airborne radioactive material concentration levels) will be developed for assigning respiratory protection, collecting bioassay samples, and stopping work.

- Respiratory protection shall be considered if a worker's intake is expected to exceed 40 DAC-hours in a week.
- A bioassay program must be implemented for any worker whose intake is expected to exceed 10% ALI or 40 DAC-hours in a week.
- Work shall be ceased if air sampling results show greater than 10% DAC is present. The RSO shall evaluate the situation and provide recommendations for restarting work for the approval of the EPM PM.

10.8.2 Air Sample Analysis

Air samples will be counted on-site using existing laboratory bench top scalers (e.g., Ludlum Model 3030 or similar equipment). MDAs based on various sample count times will be calculated and used to determine the sample volume needed to detect less than 10% DAC for 5% enriched uranium. This information will be documented and used to determine the minimum sampling time for GA and lapel air samplers.

10.8.3 Worker Intake Determination

To demonstrate that potential worker intakes are less than 2% ALI, the following methodology for assessing exposure to internal emitters shall be employed:[1]

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[1] Ryan, M.T., et.al., "A Simple Method for Assessing Exposure to Internal Emitters," *Health Physics Journal*, Vol. 80, suppl 2, May 2001

Calculation of expected intake from air sample results is quite simple and does not require stay time in the airborne radioactivity environment. Intake is commonly obtained by performing a bioassay measurement and dividing that value by a fraction of intake expected in the bioassay organ, tissue, or excreta called an intake retention fraction (IRF). Performing intake estimates from personal air sample data is no different. However, the intake retention fraction is derived from the breathing rate of Reference Man, F_{RM}, and the flow rate of the personal air sample, F_{PAS}.

$$IRF = \frac{F_{PAS}}{F_{RM}}$$

The intake (I) is calculated from the personal air sample activity, A_{PAS}

$$I = \frac{A_{PAS}}{IRF} = \frac{A_{PAS}}{F_{PAS}/F_{RM}}$$

Substituting Reference Man's breathing rate of 20 L m^{-1}, we obtain

$$I = \frac{20\ A_{PAS}}{F_{PAS}}$$

Dose can be inferred from this intake by applying a dose coefficient (Federal Guidance Report No. 11 published by the U.S. EPA in 1988).

The intakes determined using the last equation provided above will be used to demonstrate that potential worker intakes during the calendar year are less than 2% ALI. If the potential intake for workers exceeds 2% ALI, the RSO will perform an

investigation, including consideration of data from general area air samples to determine if internal exposure monitoring is required (see Section 6.6).

10.9 Survey Training

Surveys shall be performed by personnel who have been trained commensurate with the type of surveys to be performed. Training will address the following, as applicable:

- Appropriate instrumentation to be used,
- Operational and response checks for survey instrumentation,
- Survey methods, recording of data,
- Calculations, data evaluation, and
- Action levels.

10.10 Survey Documentation

Radiation, contamination, and airborne radioactivity surveys performed for compliance purposes, or radiation and contamination surveys performed to demonstrate that decommissioning criteria have been met, shall be documented and maintained in accordance with 10 CFR 20, Subpart L and the QAPP.

11.0 RADIOACTIVE MATERIALS CONTROL

11.1 Section Overview

This section addresses radioactive material controls employed at the Cimarron Site. This section includes requirements related to the following:

- Material Control and Accountability (MC&A)
- Receipt, Labelling, and Storage of Radioactive Material
- Shipment and Transfer of Radioactive Material
- Controls for Radioactive Sources.
- Theft or Loss of Radioactive Material

11.2 Material Control and Accountability

The potential for a nuclear criticality event during the proposed decommissioning program at the Cimarron Site is extremely unlikely because both the concentration and the enrichment of uranium in material generated during decommissioning are low. Treatment of groundwater to remove the enriched uranium from groundwater will result in a more concentrated form of uranium on the ion-exchange resin. The accumulation of enriched uranium on resin has been evaluated by an analysis to demonstrate nuclear criticality safety. The purpose of the MC&A Program is to ensure that the accumulation of enriched uranium during groundwater processing does not exceed the license limits for possession of U-235 (i.e., 1,200 grams of U-235) and to ensure that packaged radioactive waste meets all regulatory requirements for transportation.

11.2.1 Responsibilities and Training

11.2.1.1 The RSO is responsible for evaluating proposed changes to the decommissioning facilities and processes in consultation with an individual

with experience in nuclear criticality safety evaluation. The RSO will review and approve any changes made to decommissioning systems and/or operations and will periodically conduct inspections to confirm that engineering and administrative controls prevent exceeding license possession limits.

All personnel responsible for operations will receive training on the potential for nuclear criticality and the need to comply with the controls established to maintain nuclear criticality safety during treatment and processing operations.

11.2.1.2 Training

(a) MC&A training will address: Awareness of the significance of exceeding the basic parameters necessary to stay within the Nuclear Criticality Safety analysis which are:

- Any measurement of an enrichment >7.33% U-235,
- Any measurement of the U-235 concentration on the resin >8g/kg,
- The need to assure that the U-235 concentration in packaged waste containers is <0.5g U-235 per kg of waste, and
- Any change in the storage of containers or process equipment that would result in a height >7 feet.

(b) Awareness of the need to assure that the process system inventory does not exceed 1,200 grams of U-235,

(c) Awareness of the need to assure that the total Site inventory does not exceed 0.5 effective kilograms of U-235 to ensure certain regulatory requirements, such as stringent security of SNM, are not invoked.

(d) Awareness that any measurement of an enrichment >5% requires downgrading action in accordance with the license possession limits.

(e) Awareness that all individuals are required to implement an immediate "stop work" response if any of the above listed parameters are violated.

11.2.2 Controls

Controls are addressed in MC&A procedures. If any of the basic parameters are exceeded, the nuclear criticality safety analysis has been invalidated and it would be necessary to stop processing operations until either the analysis is redone and/or the situation corrected that led to the exceedance. Controls are implemented in MC&A procedures which are based on the following assumption/practices:

11.2.2.1 Uranium Mass Determination

Uranium mass determinations will be based on analytical measurements using the ICP-MS (EPA 200.8) method to report the U-235 and U-238 mass concentrations.

11.2.2.2 Enrichment Value Calculation

The enrichment values will be calculated using analytical measurements of U-235 and U-238. The enrichment of the uranium is calculated (ignoring the U-234 mass contribution) by:

$E = M_{U\text{-}235} / (M_{U\text{-}235} + M_{U\text{-}238})$

Where:

E = the Uranium enrichment level in wt. % U-235

M_i = the mass of the isotope in micrograms isotope per liter or gram of sample

11.2.2.3 Site Inventory

The Site inventory of U-235 will be tracked using two logs: the *Mass Inventory Log* and the *Container Inventory Log*. The *Mass Inventory Log* tracks the U-235 content of the processing system including well field piping, tankage, dual-media filters, resin vessels, resin blending equipment, and waste containers that have not been determined to meet the transportation requirements. The U-235 content of the groundwater that has not been extracted from the ground is not included in the Site inventory. Once a waste container (spent resin or other solid waste) is prepared for storage pending transportation to disposal, The U-235 content of the container is entered on the *Container Inventory Log* and any associated inventory values on the *Mass Inventory Log* are set to zero. The total U-235 inventory of both the *Mass Inventory Log* and the *Container Inventory Log* represents the total Site inventory of U-235, which is only used to demonstrate that the total Site inventory does not exceed 0.5 effective kilograms of U-235.

11.2.2.4 Process Line, Tank, and Filter Inventory

SNM mass contents of process lines transporting and influent tanks storing groundwater, and dual-media filters will be conservatively estimated based on reasonable concentrations and enrichments of the uranium. These components will contain minor quantities of SNM. These conservatively established mass values may be re-evaluated and revised as appropriate if information is

obtained that the difference in calculated mass of U-235 is significant to the mass inventory value.

11.2.2.5 Resin Inventory

The SNM contents of resin vessels will be established based on the total flow of groundwater through the vessel and the difference between the input and output uranium concentrations of the flow. The enrichment of the SNM will be initially based on conservatively established values until analytical results for spent resins provide actual enrichment values. The following points describe in greater detail the process for establishing the total U-235 inventory for the resin material:

11.2.2.6 During the initial system startup phase the sample analysis turn-around time will be reduced to obtain data on an expedited basis

11.2.2.7 Water samples for each treatment train will be taken from the influent and the effluent from each of the lead, lag and polish vessels

11.2.2.8 The enrichment of the uranium for each train will initially be the assumed values presented in Appendix N of the D-Plan and will be revised when analytical results are obtained from samples of processed resin

11.2.2.9 The mass of U-235 added to each vessel will be calculated based on the total flow of water processed through the vessel for each time-period between sampling events times the difference between the average influent and effluent water concentration for that vessel. The total inventory of the vessel is the sum of all mass determinations over the entire period the vessel has been in the treatment process

11.2.2.10 The total U-235 inventory of the resin material is the sum of the U-235 contained in:

- all the resin vessels in the treatment systems,
- vessels containing spent resin if any are being stored while awaiting the blending process,
- resin in the blending equipment, and
- packages of processed resin that have not been transferred to storage

11.2.2.11 Once a vessel has been emptied of resin the inventory value for that vessel will be set to zero.

11.2.2.12 The minimum quantity of absorbent material to be blended with the resin to yield a fissile exempt, dry resin mixture will be calculated based on assumed enrichment and the uranium mass derived from process measurements (with safety factor).

11.2.2.13 The resin from the lead vessel changeout will be blended with absorbent and loaded into waste packages.

11.2.2.14 The sampling of the processed resin/absorbent waste will be performed in accordance with an approved sampling procedure.

11.2.2.15 Initially the SNM content of packages of waste will be based on process measurements. Upon receipt and validation of analytical data the SNM content of the packaged waste material will be finalized, the *Container Inventory Log* updated, and the packages will be relocated to the "ready-to-ship" area of the storage area.

11.2.2.16 Waste containers which are awaiting final analytical data will be stored in an "in-process" area within the WATF.

11.2.2.17 Container Inventory

Once a package of waste has been determined to meet the transportation and waste disposal requirements, the SNM contents of the package will be removed from the *Mass Inventory Log* and the package and SNM contents will be entered on the *Container Inventory Log.*

11.2.2.18 Removal From Site Inventory

When a package is shipped to disposal the package and SNM contents will be removed from the *Container Inventory Log.*

11.3 Receipt, Labeling, and Storage of Radioactive Material

Any radioactive materials received at the Site shall be received in accordance with radioactive material license possession limits and 10 CFR 70.19. The individual responsible for radioactive material receipt shall ensure that all surveys required by 10 CFR 20.1906 are performed and review shipment paperwork to ensure compliance with 49 CFR.

Each container of radioactive material stored on-site shall be labeled as required by 10 CFR 20.1904.

Radioactive material shall be secured against unauthorized access or removal. Radioactive material storage areas shall be posted and controlled using appropriate barriers and radiological postings.

11.4 Shipment and Transfer of Radioactive Material

Radioactive material shipments shall comply with NRC (10 CFR) and U.S. Department of Transportation (49 CFR) regulations. Low-level radioactive waste shipments transferred for disposal shall be accompanied by a shipment manifest prepared in accordance with 10 CFR 20.2006. Radioactive material shall only be transferred to authorized individuals in accordance with the appropriate regulations in 10 CFR 20, and 10 CFR 70.

Transfer of radioactive material between RAs is addressed procedurally to ensure contamination controls discussed in Section 12 are satisfied and that any personnel exposures are ALARA.

11.5 Controls for Radioactive Sources

The RSO shall approve all requisitions for radioactive sources and ensure that source inventories are performed on a quarterly basis. Radioactive sources shall be tested for leakage and/or contamination upon receipt and on a quarterly basis, except that any licensed sealed source is exempt from leak tests if the source contains less than 0.1 microcuries of plutonium or uranium, 100 microcuries of beta and/or gamma emitting radioactive material or 10 microcuries of other alpha emitting radioactive material. Leak testing and inventory of Exempt Quantity radioactive sources is not required; however, these sources are stored in a secure area to prevent unauthorized removal or access.

Unless specifically authorized by the RSO, electroplated sources are not smear tested for leakage to prevent removal of radioactive material from the electroplating.

The RSO shall approve locations for storage of radioactive sources. Radioactive source storage areas shall be secured against unauthorized removal or access of licensed radioactive material and posted per 10 CFR 20.1902.

11.6 Theft or Loss of Radioactive Material

Any individual who discovers that radioactive material is lost, stolen, or missing shall immediately notify the RSO. The RSO shall evaluate the physical and radiological characteristics of the missing material and the potential hazards to workers and the general public, initiate an investigation to locate the material, and perform a root cause evaluation of the incident. The RSO shall determine the need for notifications to regulatory authorities and make notifications as necessary per 10 CFR 20.2201.

12.0 CONTAMINATION CONTROL

12.1 Section Overview

The purpose of contamination control is to prevent and/or minimize the spread of radioactive contamination to individuals, areas, and equipment. Control of radioactive surface contamination prevents or minimizes possible inhalation or ingestion of radioactive material by personnel, skin dose from small particles of radioactivity, and the spread to or build-up of radioactive material in the facility or environment from decommissioning operations. Controls to prevent the spread of contamination shall be proposed by the Activity Leaders and approved by the RSO or designee prior to implementation.

12.2 Contaminated Buildings and Equipment

Radioactive contamination of buildings and equipment located within an RA shall be maintained below the removable contamination limit of 1,000 dpm/100 cm^2 alpha. In addition, Contaminated Area controls, including posting, shall be implemented whenever removable contamination in an Unrestricted Area exceeds 1,000 dpm/100 cm^2 alpha or 1,000 dpm/100 cm^2 beta-gamma. The Site incorporates the ALARA philosophy when selecting decontamination methods and practices.

As a general rule, decontamination is performed by working from areas of low contamination to areas of high contamination if possible. Decontamination materials should be limited to the minimum required for the task. All decontamination materials shall be collected, monitored, and properly dispositioned. Table 12-1 provides a summary of contamination action levels and associated actions to be taken.

12.3 Contaminated Personnel

Decontamination of personnel shall be performed under the guidance of health physics personnel and shall incorporate good health physics practices and the ALARA principle. An individual whose skin or personal clothing is found contaminated above background shall not exit an RA without prior approval of the RSO. Appropriate surveys and monitoring shall be performed to evaluate dose to the individual resulting from contamination.

12.4 Spill of Radioactive Material

A spill of radioactive material requires immediate actions which include:

- Stop the spill
- Warn other personnel
- Isolate the area
- Minimize radiation exposure
- Secure the area and stand guard (until otherwise direct by health physics personnel)

Supplementary actions should include the performance of radiological surveys in immediate and adjacent areas, including downwind.

12.5 Contamination Control During Groundwater Processing

Contamination control during groundwater processing involves both process operations and activities necessary to supply groundwater to the processing facility. This section of the RPP is intended to implement contamination control commitments identified in the D-Plan and supplements section 8.2 of the RPP.

12.5.1 Determination of Background Levels

Section 11.6 of the D-Plan discusses the need for describing surveys that will be performed to determine the baseline of background radiation levels and radioactivity from natural sources for areas where decommissioning activities will take place. The following discussion addresses methods for determining background/baseline radiation and radioactivity levels during decommissioning activities consistent with information provided in section 3.6 of the D-Plan. These methods may be modified or changed by the RSO subject to ALARA Committee review.

12.5.1.1 Background Radiation Exposure Rate Levels

As discussed in section 3.6.1 of the D-Plan, background exposure rates have historically ranged from 7 to 10 μR/hr. These values have been previously reviewed and accepted by the NRC. More recent surveys, expand this range to 6 to 11 μR/hr. For decommissioning purposes, the background exposure rate will be measured in an unaffected area with a similar soil/surface on the day the survey is conducted. Any background exceeding 11 μR/hr will be evaluated by the RSO or designee and the background exposure rate value for that survey documented in the survey report.

12.5.1.2 Background Levels of Uranium in Soil

As discussed in section 3.6.1 of the D-Plan, the NRC agreed that if alpha spectroscopy is used to analyze samples, a value of 2.8 pCi/g total uranium is acceptable for use as the background activity concentration for soil. This value will apply site-wide. When soil is analyzed for mass concentration, analysis will be for U-235 and U-238 (the mass of U-234 is negligible). A U-235

enrichment of [U-235/(U-235 + U-238)] will be used to convert the mass concentration for U-234, U-235, and U-238 to activity concentration for comparison with decommissioning criteria.

12.5.1.3 Background Levels of Uranium in Groundwater

Information on the background concentration of uranium in groundwater is provided in section 3.6.4 of the D-Plan for information only. Unlike the decommissioning criteria for soil or surface contamination, the decommissioning criteria for uranium in groundwater is not a "concentration above background". Consequently, there is no reason to establish a background value to which the NRC Criterion of 180 pCi/L total uranium would be added.

12.5.1.4 Background Levels of Uranium in Surface Water

For surface water, background concentration of uranium will be determined for each sampling event by using the results from the upstream sample collected from the Cimarron River. As discussed in section 3.6.3 of the D-Plan, the isotopic activity concentration data from 26 sampling events at the upstream sampling location were tabulated. The mean total uranium concentration for this sampling location is 4.8 pCi/L; the maximum total uranium concentration obtained from that location is 8.8 pCi/L.

12.5.2 Soil Disturbances

Subsurface soil will be brought to the surface during installation of injection and extraction trenches, monitoring wells, trenches for piping and utilities, etc. As provided in the D-Plan Section 3.3, both surface and subsurface soil has been demonstrated to

comply with NRC Criteria Site-wide. The subsections below discuss requirements for surveying and sampling subsurface material at the Site. Results will demonstrate that the subsurface materials contain contamination less than or equal a net 30 pCi/g uranium. Walk-over gamma surveys are used to determine relative levels of radioactivity (counts per minute) using a gamma sensitive detection instrument, such as a 2-inch by 2-inch NaI detector. Instrument selection will be approved by the RSO and identified in the applicable desk instruction or activity plan.

12.5.2.1 Foreign or Suspect Materials

Gamma surveys will be performed should soil-like material (e.g., crushed asphalt, debris, etc.) which may be volumetrically contaminated be encountered during excavation of trenches for piping, instrumentation, foundations, etc. Soil-like material yielding elevated gamma counts per minute will be removed, segregated, and sampled for laboratory analysis. If not releasable, the material will be packaged and disposed of as low-level radioactive waste.

12.5.2.2 Shallow Excavations

Gamma walk-over surveys shall be performed of subsurface soil from trenches excavated for installation of instrumentation, piping, and utilities, and other excavations 4 feet or less in depth. Instructions for performing and documenting the survey shall be documented in a desk instruction or activity plan to include the following:

- A gamma walk-over survey shall be performed over the length of the spoils. A zig-zag pattern should be employed.

- A soil sample shall be taken once every 100 feet along the length of the spoils at the location with the highest count rate. If a count rate exceeds two times background, the area will be identified (e.g., flagging or staking) and the RSO will evaluate the survey results and determine if additional sampling should be collected for laboratory analysis.

12.5.2.3 Deep Excavations

Gamma walk-over surveys shall be performed of subsurface soil from the installation of extraction and injection trenches and other excavations that are greater than 4 feet in depth. Instructions for performing and documenting the survey shall be documented in a desk instruction or AP to include the following:

- A gamma walk-over survey shall be performed over the length of the spoils. A zig-zag pattern should be employed.
- A soil sample shall be taken once every 100 feet along the length of the spoils at the location with the highest count rate. If a count rate exceeds two times background, the area will be identified (e.g., flagging or staking) and the RSO will evaluate the survey results and determine if additional sampling should be collected for laboratory analysis.

12.5.2.4 Borings and Wells

Borings for installation of monitoring and extraction wells will yield soil cores varying from less than two feet to five feet in length. Each increment shall be

surveyed using a sensitive gamma detector (e.g,. 2 inch by 2 inch NaI detector or alternate approved by the RSO). A minimum of one soil sample shall be collected from each five-foot depth interval, at the location where the highest gamma count rate is detected, for laboratory analysis.

12.5.2.5 WATF Disturbed Area

Grading required for construction of the WATF will involve excavation of soils and back-fill with clean imported material. Surveys and soil sampling are not required in areas where clean back-fill material is placed. Consistent with the methodology discussed in NUREG/CR-5849 for affected areas, soil sampling shall be performed in areas where soil is excavated below grade following completion of construction activities. A 10-meter grid will be established and a gamma walk-over survey will be employed to collect soil samples shall be taken at a depth of 0 to 6 inches at the highest radioactivity location that yields the highest gamma counts per minute in each grid.

12.5.2.6 BARF Excavation

Grading required for construction of the BARF will involve cut and fill, in which subsurface soil will be exposed at final grade in some areas, and excavated soil will be placed over what had been surface soil in other areas. Surveys and soil sampling shall be performed following completion of construction activities. Consistent with the methodology for affected areas provided in NUREG/CR-5849, a 10-meter grid will be established and a gamma walk-over survey will be employed to collect soil sample at a depth of

0 to 6 inchess at the location that yields the highest gamma counts per minute in each grid.

12.5.2.7 1206 Sediment/Soils Mixture

As discussed in section 8.2.4 (1206-North) of the D-Plan, the 1206 Drainage is unique in that it is the only area in which excavation and disposition of sediment will be performed as a groundwater remediation strategy. The sediment will be mixed with excess spoils generated during trench excavation and placed in a soil laydown area. Following mixing and placement, vegetation will be established over the material.

After placement of the sediment/spoils mixture is complete, a 10-meter grid will be established over the laydown area. Samples of the mixture will be collected at each grid location. Composite samples representing each one-foot depth interval will be collected from each location. Duplicate samples will be collected at a minimum of 10% of the 10-meter grid locations.

Samples collected from the 10-meter grid locations will be submitted for isotopic analysis. An additional set of 20 "confirmatory" samples will be collected from randomly selected 10-meter grid locations. These samples will be retained should the NRC desire to analyze those samples for confirmatory survey or inter-laboratory comparison.

For each 10-meter sample that yields less than 30 pCi/g total uranium above background, the material within that grid will be considered in compliance with the NRC license criterion. Should a sample from a 10-meter grid location exceed 30 pCi/g above background, samples of the sediment/spoils

mixture will be collected from the same depth interval on a 5-meter grid surrounding that sample to demonstrate that the average activity over a 100 m^2 area complies with the license criterion. If the average activity for the 10-meter grid sample and the four surrounding 5-meter grid locations is less than the license criterion, the material within that grid will be considered in compliance with the license criterion.

Should any grid/depth interval exceed the license criterion, that material will be excavated, placed in drums along with sufficient absorbent to ensure that there will be no free liquid and disposed of as low-level radioactive waste.

12.5.3 Facility Operations

Low-enriched uranium will be processed through ion exchange resins that will concentrate the uranium in the resins. The concentration of uranium on these resins represents a source of potential contamination. The following contamination controls ensure that contamination is contained and not spread throughout the processing facilities or across the Site.

12.5.3.1 Influent piping contains low concentrations of uranium with little potential for generating contamination. Routine monitoring is performed during operations to ensure that contamination is controlled and not being spread at well heads where the groundwater is extracted. Connections to the water treatment systems are inspected and monitored to identify and repair leaks.

12.5.3.2 Engineering controls are included in the design of the groundwater treatment system to contain contamination during ground water processing.

Double walled tanks are used to contain the influent groundwater awaiting processing. Ion exchange resins are contained in stainless steel vessels. Spent resins are processed through a wet process that ensures airborne radioactivity is not generated. The spent resin is processed in an enclosed system. Spent resin is packaged as discussed in the D-Plan. Procedures for contamination monitoring and air sampling are provided to demonstrate the effectiveness of these engineering controls.

12.5.3.3 Engineering controls are incorporated in the design to eliminate or minimize the potential for drips and leaks during sampling, resin vessel changeout, and spent resin processing. Protective clothing shall be prescribed for maintenance activities involving potential exposure to spent resins.

12.5.3.4 Effluent from treatment systems must contain uranium at concentrations below drinking water standards, as demonstrated by discharge sampling requirements specified in the discharge permit issued by the Oklahoma Department of Environmental Quality. Leaks or unintentional releases of effluent do not constitute contamination control concerns.

12.5.3.5 The QAPP requires that only appropriately trained workers (Section 2.3) are permitted access to Contaminated Areas. Work performed in Contamination Areas will be performed in accordance with procedures that require measures incorporated into the design to prevent or contain drips, leaks, etc. are correctly implemented.

Table 12-1

Contamination Action Levels

Location/Type of Contamination	Contamination Action Level	Actions to be Taken	Radiological Monitoring
Unrestricted Area – Removable Contamination	1,000 dpm/100 cm^2 beta/gamma or 1,000 dpm/100 cm^2 alpha	1. Post area/restrict access. 2. Investigate for spread and determine personnel affected. 3. Decontaminate the area and de-post, as appropriate. 4. Determine corrective actions to prevent recurrence, if necessary.	RSO or designee determines the need for increased frequency of contamination surveys.
Unrestricted Area – Fixed Contamination	1,000 dpm/100 cm^2 beta/gamma or 1,000 dpm/100 cm^2 alpha	1. Post area/restrict access. 2. Investigate cause and corrective actions. 3. Determine whether to decontaminate or implement controls to prevent spreading contamination.	RSO or designee determines the need for increased frequency of contamination surveys.
Restricted Area – Removable Contamination	1,000 dpm/100 cm^2 beta/gamma or 1,000 dpm/100 cm^2 alpha	1. Post area, if not already posted. 2. Determine source of contamination, if area was not posted and actions necessary to prevent further contamination spread. 3. Decontaminate, if appropriate.	RSO or designee determines the need for increased frequency of contamination surveys.

Table 12-1

Contamination Action Levels

Location/Type of Contamination	Contamination Action Level	Actions to be Taken	Radiological Monitoring
Restricted Area – Fixed Contamination	1,000 dpm/100 cm^2 beta/gamma or 1,000 dpm/100 cm^2 alpha	1. Post area, if not posted. 2. Determine the source of contamination and necessary actions to prevent further contamination spread. 3. Decontaminate, if appropriate.	RSO or designee determines the need for increased frequency of contamination surveys.
Release of materials for unrestricted use	See Section 13.3	1. Decontaminate or dispose of as radioactive waste.	See Section 13.3
Personnel/clothing contamination	Detectable contamination (e.g., 2 times background) on clothing or skin	1. Decontaminate personnel in accordance with Section 12.3. 2. Decontaminate or discard contaminated personal clothing if unrestricted release criteria cannot be satisfied	1. See Section 12.3 regarding personnel contamination monitoring. 2. RSO or designee authorizes release of personal clothing, if appropriate.

13.0 RELEASE FOR UNRESTRICTED USE OF MATERIALS

13.1 Section Overview

Site personnel are authorized to release tools, equipment, parts, and materials for unrestricted use provided that radiation levels and surface contamination levels do not exceed the limits in Condition 27(c) of the license. Such surveys will be performed and documented by qualified individuals.

Tools, equipment, parts, and material that do not come into contact with subsurface soil or groundwater containing radioactive material above the unrestricted release criteria do not require surveys prior to release from the Site.

13.2 Survey Instrumentation

Instruments used to perform surveys for release for unrestricted use are calibrated as discussed in Section 7.

13.3 Release Survey of Materials

As provided in License Condition 27(c), the unrestricted release criteria listed in the August 1987 "Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of License for Byproduct, Source or Special Nuclear Material" apply to the surfaces of buildings and equipment, and the October 23, 1981, BTP "Disposal or Onsite Storage of Thorium or Uranium Wastes from Past Operations," apply to soils or soil-like material. The specific values are listed in paragraphs 13.3.1 and 13.3.2.

13.3.1 Surfaces of Buildings, Materials, and Equipment

Release surveys will consist of direct (fixed + removable) and removable (smears) contamination monitoring. The Site is authorized to release materials provided that the direct and removable levels do not exceed the limits stated in the Trust license and summarized below. Such surveys will be performed and documented by qualified individuals.

Survey plans may be developed for the release of property or equipment associated with non-routine activities. Such survey plans will include the methods used to estimate uncertainty bounds for each type of instrument measurement consistent with the methodology provided in NUREG/CR-5849 section 8.2.

The release criteria for the surfaces of buildings and equipment are:

- 5,000 dpm alpha/100 cm^2 averaged over 1 m^2 (direct)
- 5,000 dpm beta-gamma/100 cm^2 averaged over 1 m^2 (direct)
- 15,000 dpm alpha/100 cm^2 maximum over 1 m^2 (direct)
- 15,000 dpm beta-gamma/100 cm^2 maximum over 1 m^2 (direct)
- 1,000 dpm alpha/100 cm^2 averaged over 1 m^2 (removable)
- 1,000 dpm beta-gamma/100 cm^2 averaged over 1 m^2 (removable)
- 5 μR/hr – above background at 1 meter

13.3.2 Soil-Like Material

Field surveys of soil-like material that may be encountered during installation of injection and extraction trenches, monitoring wells, trenches for piping and utilities, and other soil disturbances cannot demonstrate compliance with the criteria provided in License

Condition 27(c), as discussed in Section 12.5.1. Accordingly, samples of contaminated soil-like material will be submitted for laboratory analysis and segregated from other materials until analytical data is received.

14.0 RESPIRATORY PROTECTION

14.1 Section Overview

The need for a respiratory protection for radiological work is not envisioned at the Cimarron Site. Work activities that could potentially expose workers to airborne radioactive material have been evaluated to determine the potential intakes during groundwater treatment and spent resin processing. The evaluation employed the methods discussed in Regulatory Guide 8.25, Rev. 1, "Air Sampling in the Workplace" and NUREG-1400, "Air Sampling in the Workplace." If processes or operations change, then a re-evaluation of potential intakes shall be performed to determine the potential intake that could result from these changes. If the potential intake determined from this evaluation is 2% ALI or greater, the RSO will perform an ALARA evaluation when it is not practical to apply engineering controls or procedures that demonstrates that the use of respiratory protection equipment is ALARA. If the ALARA evaluation demonstrates that use of respiratory protection equipment is ALARA, then the RSO will implement the respiratory protection program described in this section.

Respiratory protection measures shall be employed when necessary to protect workers from airborne hazards. Groundwater treatment results in the generation of moist treatment media with little potential to generate airborne radioactivity. However, if future conditions change and the RSO or designee determines, through review of field conditions or anticipated work functions, that respiratory protection is required, procedures and controls will be instituted in accordance with the requirements found in 10 CFR 20, Subpart H, "Respiratory Protection and Controls to Restrict Internal Exposure in Restricted Areas" for radiological hazards and the 29 CFR 1910.134 for non-radiological hazards. Section 14.2 provides specific requirements for the respiratory protection program, if needed.

If a respiratory protection program is determined to be necessary, the program will be based on guidance provided in Regulatory Guide 8.15, Rev. 1, "Acceptable Programs for Respiratory Protection," and NUREG/CR-0041, Rev. 1, "Manual of Respiratory Protection Against Airborne Radioactive Material."

14.2 Respiratory Protection Program

The need for respiratory protection for radiological work is not envisioned at the Site. However, respiratory protection will be required if work activities could potentially expose workers to 40 or more DAC-hours in a week. Respiratory protection will also be required for any areas where airborne radioactive material concentrations are expected to exceed 1 DAC. If either of these trigger levels are encountered, as required by 10 CFR 20.1703(c), the respiratory protection program will include the following elements outlined in Section 11.2 of the D-Plan:

- A description of the process controls, engineering controls, or procedures to control concentrations of radioactive materials in air.
- A description of the evaluation that will be performed when it is not practical to apply engineering controls or procedures, that demonstrates that the use of respiratory protection equipment is ALARA.
- A description of the considerations used to demonstrate that respiratory protection equipment is appropriate for a specific task, based on the guidance on assigned protection factors (APF)
- A description of the medical screening and fit testing required before workers will use any respirator that is assigned a protection factor.

- A description of the written procedures maintained to address all the elements of the respiratory protection program.
- A description of the use, maintenance, and storage of respiratory protection devices in such a manner that they are not modified and are in like-new condition at the time of issue.
- A description of the respiratory equipment users' training program.
- A description of the considerations made when selecting respiratory protection equipment to mitigate existing chemical or other respiratory hazards instead of (or in addition to) radioactive hazards.

The respiratory program documentation will include the following details:

- Air sampling sufficient to identify the potential hazard, permit proper equipment selection, and estimate doses;
- Surveys and bioassays, as necessary, to evaluate actual intakes;
- Testing of respirators for operability (user seal check for face sealing devices and functional check for others) immediately prior to each use;
- Written procedures regarding—
  - Monitoring, including air sampling and bioassays;
  - Supervision and training of respirator users;
  - Fit testing;
  - Respirator selection;
  - Breathing air quality;
  - Inventory and control;

  - Storage, issuance, maintenance, repair, testing, and quality assurance of respiratory protection equipment;
  - Recordkeeping; and
  - Limitations on periods of respirator use and relief from respirator use.
- Determination by a physician that the individual user is medically fit to use respiratory protection equipment:
  - Before the initial fitting of a face sealing respirator;
  - Before the first field use of non-face sealing respirators, and
  - Either every 12 months thereafter, or periodically at a frequency determined by a physician.
- Fit testing, with fit factor > 10 times the APF for negative pressure devices, and a fit factor > 500 for any positive pressure, continuous flow, and pressure-demand devices, before the first field use of tight fitting, face-sealing respirators and periodically thereafter at a frequency not to exceed 1 year. Fit testing must be performed with the facepiece operating in the negative pressure mode.

15.0 ENVIRONMENTAL MONITORING

15.1 Section Overview

Environmental monitoring shall be performed at various locations to monitor the migration of licensed material from former (now decommissioned) sources through environmental media. Final surveys have demonstrated that buildings and soils have been decommissioned. Licensed material exceeds decommissioning criteria in groundwater in three areas: Burial Area #1, the Western Upland Area, and the Western Alluvial Area. The Licensee shall maintain an environmental monitoring program in these three areas until superseded by a groundwater remediation work plan.

Effluent from the groundwater treatment process will be monitored to demonstrate that the concentrations of uranium complies with discharge permit limits and underground injection permits. Monitoring will be performed in accordance with permit requirements and the Sampling and Analysis Plan. The Sampling and Analysis Plan establishes procedures for the collection, evaluation, and analysis of samples. A radiation protection procedure will address how background and baseline concentrations of radionuclides ineffluents are established through appropriate sampling and analysis. The procedure will include the following information:

- A description of known or expected concentrations of radionuclides in effluents;
- A description of the physical and chemical characteristics of radionuclides in effluents;
- A summary or diagram of all effluent locations;
- Justification that samples are representative of actual releases;
- A summary of the sample collection and analysis procedures, including the minimum detectable concentrations of radionuclides;

- A summary of sample collection frequencies; and
- A description of effluent release monitoring recording and reporting procedures.

Direct radiation from groundwater processing operations is monitored in the vicinity of the Western Area Treatment Facility and Burial Area #1 Remediation Facility, as discussed in Section 15.3.

15.2 Surface and Groundwater Monitoring

Surface and groundwater samples are collected annually and are analyzed for fluoride, nitrates/nitrites, gross alpha radioactivity, gross beta radioactivity, and uranium isotopes. The locations identified in Table 15-1 shall be sampled on an annual basis.

Upon approval of D-Plan, the in-process groundwater monitoring plan described in Sections 8.6 and 12.2 of the D-Plan will replace the environmental monitoring program described in the preceding paragraph.

15.2.1 Quality Control in Sampling

Sample collection, preservation, shipping, and analysis shall be conducted in accordance with the Sampling and Analysis Plan and associated procedures. Data review, reporting, and management will be conducted in accordance with Quality Assurance Implementing Procedure, QAIP-17.1, "Data Management Procedure."

15.2.2 Reporting

Environmental monitoring results shall be reported to NRC within 30 days of the completion of data review.

15.3 Direct Radiation

Dosimeters were deployed on October 1, 2019. These dosimeters collected background dose in the vicinity of the Western Area Treatment Facility and Burial Area #1 Treatment facility prior to construction of these facilities. These dosimeters are used to establish the baseline of background radiation levels prior to commencing decommissioning activities. Additionally, one dosimeter was deployed along the haul path between the facilities. Effective October 1, 2022, dosimeters in Burial Area #1 and along the haul path were discontinued because the proposed treatment facility at Burial Area #1 was removed from the D-Plan. Once decommissioning activities commence, dosimeters will be used to determine radiation levels outside the RA from groundwater processing activities. Figure 15-1 depicts dosimeter locations. Table 15-2 provides a verbal description of the dosimeter locations. Dosimeters are exchanged on a quarterly basis.

NOTE: Dosimeter locations will be reevaluated during construction of these facilities and adjusted, if appropriate, by the RSO. The rationale for such adjustments shall be documented. Additionally, dosimeter locations will be periodically evaluated during decommissioning activities to determine if locations need to be adjusted or removed. Justification for changes shall be documented and approved by the RSO.

15.4 Quality Assurance Program

QAIP-17.1, *Data Management Practice* implements the quality assurance for the effluent monitoring program at the Site. QAIP-17.1 addresses the following aspects of this program:

- Electronic data deliverable formats.
- Data review parameters and methods.
- Archiving data as quality records.

The procedure applies to analytical data supplied by contract laboratories, geodetic data collected by licensed surveyors, and field measurements as applicable to the effluent monitoring and sampling program.

Table 15-1

Surface and Groundwater Monitoring Locations

BURIAL AREA #1	WESTERN UPLAND AREA
1314	1351
TMW-08	1352
TMW-09	1354
TMW-13	1356
02W06	
02W08	
02W09	**WESTERN ALLUVIAL AREA**
02W16	MWWA03
02W17	MWWA09
02W27	T-62
02W28	T-64
02W32	T-70R
02W35	T-76
02W42	T-77
02W43	T-79
02W44	T-82
SURFACE WATER	
1201 Cimarron River Upstream	
1202 Cimarron River Downstream	

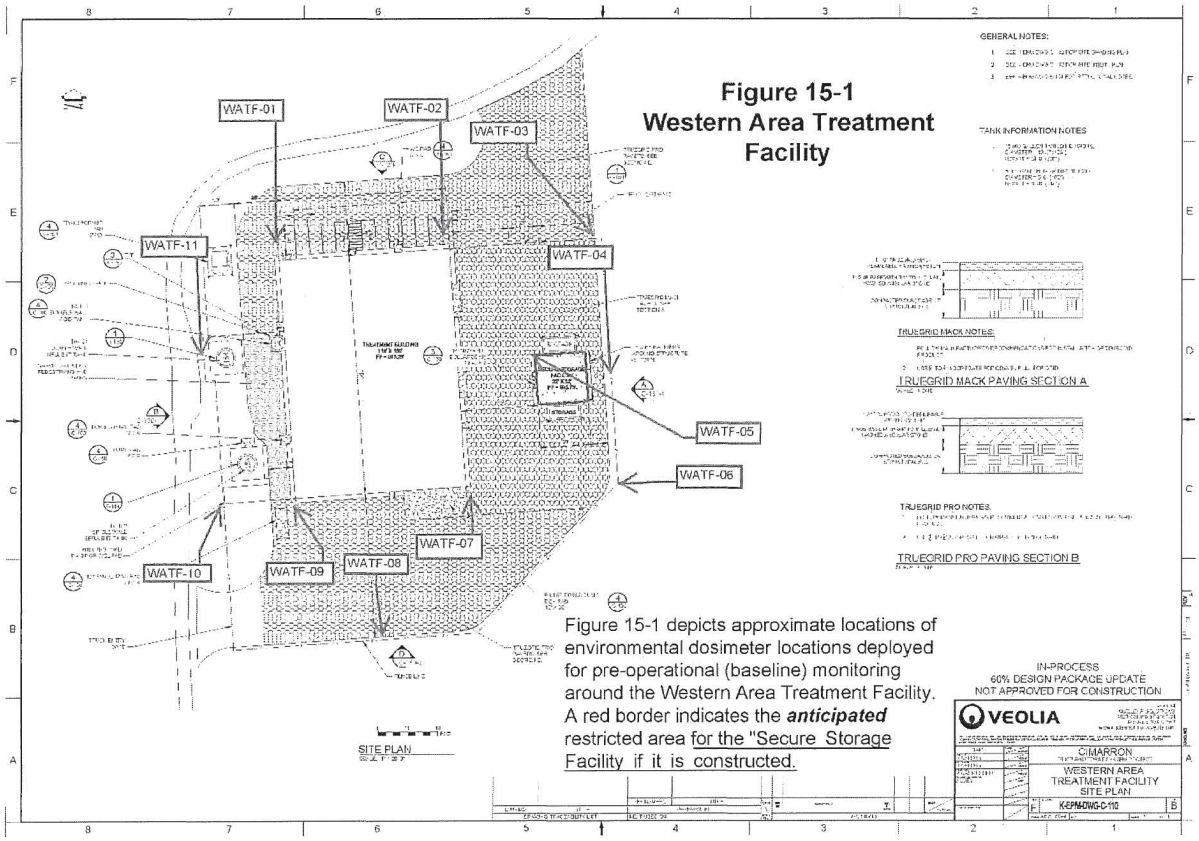
Table 15-2

Environmental Dosimeter Locations

Location Designation	Description
Western Area Treatment Facility	
WATF-01	Northwest corner of Treatment Building.
WATF-02	Northeast corner of Treatment Building.
WATF-03	Eastern fence line in line with the northern wall of the Treatment Building.
WATF-04	Eastern fence line approximately center of the Secure Storage Facility.
WATF-05	Northwest corner of the Secure Storage Facility.
WATF-06	To the southeast of the Treatment Building along the eastern fence line where the fence runs to the southwest.
WATF-07	Southeast corner of Treatment Building.
WATF-08	South fence line at Treatment Building mid-point.
WATF-09	Southwest corner of Treatment Building.
WATF-10	Point directly west of the southwest corner of the Treatment Building on the western fence line.
WATF-11	Point directly west of the centerline of the Treatment Building on the western fence line.

Figure 15-1

Western Area Treatment Facility



16.0 DEFINITIONS

Absorbed Dose: Energy imparted by ionizing radiation per unit mass of irradiated material. The units of absorbed dose are the rad and the gray (Gy). 1 Gy = 100 rad.

Adult: An individual 18 or more years of age.

Airborne Radioactive Material or Airborne Radioactivity: Radioactive material dispersed in the air in the form of dusts, fumes, particulates, mists, vapors, or gases.

Airborne Radioactive Material Area or Airborne Radioactivity Area: A room, enclosure, or area in which airborne radioactive materials, composed wholly or partly of licensed material, exists in concentrations:

(1) in excess of the derived air concentrations (DAC) specified in appendix B of 10 CFR 20.1001 – 20.2401, or

(2) to such a degree that an individual present in the area without respiratory protection equipment could exceed, during the hours an individual is present in a week, an intake of 0.6% of the Annual Limit on Intake (ALI) or 12 DAC hours.

ALARA: An acronym for "As Low As is Reasonably Achievable". ALARA is an important principle for any worker, exposed to radiation, to fully understand and apply in everyday use. The ALARA principle means making every reasonable effort to maintain exposures to radiation as far below the dose limits in 10 CFR 20 as is practical consistent with the purpose for which the licensed activity is undertaken, taking into account the state of technology, the economics of improvements in relation to state of technology, the economics of improvements in relation to benefits to the public health and safety, and other societal and socioeconomic considerations, and in relation to utilization of nuclear energy and licensed materials in the public interest.

ALARA Committee: The Cimarron Site ALARA Committee that has responsibility for overall coordination of the ALARA Program. The Committee is composed of members as described in Section 4.0 of this RPP and meets on a regular basis (typically, quarterly) to review the status of the ALARA Program and to approve changes to the RPP and D-Plan.

Alpha Particle: A positively charged particle ejected spontaneously from the nuclei of some radioactive elements. It is identical to a helium nucleus that has a mass number of 4 and an electrostatic charge of +2, i.e., two protons and two neutrons.

Annual Limit on Intake (ALI): The derived limit for the amount of radioactive material taken into the body of an adult worker by inhalation or ingestion in a year. ALI is the smaller value of intake of a given radionuclide in a year by the reference man that would result in a committed effective dose equivalent of 5 rems (0.05 Sv) or a committed dose equivalent of 50 rems (0.5 Sv) to any individual organ or tissue. (ALI values for intake by ingestion and by inhalation of selected radionuclides are given in Table 1, Columns 1 and 2, of appendix B to 10 CFR 20.1001 thru 20.2401).

Audit: An audit is an evidence gathering process. Audit evidence is used to evaluate how well audit criteria (procedures, requirements, policies) are being met. Audit evidence is used to determine how well policies are being implemented, how well procedures are being applied, and how well requirements are being met.

Atomic Number (Symbol Z): The number of protons in the nucleus of an atom.

Background: Ambient signal response recorded by measurement instruments that is independent of radioactivity contributed by the radionuclide being measured in the person or sample.

Background Radiation: Radiation from cosmic sources; naturally occurring radioactive material, including radon (except as a decay product of source or special nuclear material); and global fallout as it exists in the environment from the testing of nuclear explosive devices or from past nuclear accidents such as Chernobyl that contribute to background radiation and are not under the control of the Licensee. "*Background radiation*" does not include radiation from source, byproduct, or special nuclear materials regulated by the NRC.

Becquerel (Bq): The term used to describe one disintegration per second (dps).

Beta Particle: Beta particles are emitted by the nucleus of an atom to attain stability. Beta particles are usually negatively charged and are emitted from the nucleus of atoms with an excess of neutrons and serve to reduce the number of neutrons in the nucleus. Some beta particles are positively charged. These positively charged beta particles, known as positrons, are emitted from a nucleus and result in an increase in the number of neutrons in the nucleus. Negatively charged beta particles and positively charge positrons have a mass equal to 1/1837 that of a proton. Beta particles are easily stopped by a thin sheet of metal or plastic.

Bioassay (radiobioassay): The determination of kinds, quantities or concentrations and, in some cases, the locations of radioactive material in the human body, whether by direct measurement or by analysis and evaluation of materials excreted or removed from the human body.

Breathing Zone: The breathing zone is that region adjacent to a worker's mouth and nostrils from which air is drawn into the lungs while he/she is performing assigned work.

Breathing Zone Air Sample: Air which is drawn through or into the sample media and is a fair representation of the workers "Breathing Zone."

Byproduct material:

(1) Any radioactive material (except special nuclear material) yielded in, or made radioactive by, exposure to the radiation incident to the process of producing or using special nuclear material;

(2) The tailings or wastes produced by the extraction or concentration of uranium or thorium from ore processed primarily for its source material content, including discrete surface wastes resulting from uranium solution extraction processes. Underground ore bodies depleted by these solution extraction operations do not constitute "byproduct material" within this definition;

(3) (i) Any discrete source of radium-226 that is produced, extracted, or converted after extraction, before, on, or after August 8, 2005, for use for a commercial, medical, or research activity; or

(ii) Any material that—

(A) Has been made radioactive by use of a particle accelerator; and

(B) Is produced, extracted, or converted after extraction, before, on, or after August 8, 2005, for use for a commercial, medical, or research activity; and

(4) Any discrete source of naturally occurring radioactive material, other than source material, that—

(i) The NRC, in consultation with the Administrator of the Environmental Protection Agency, the Secretary of Energy, the Secretary of Homeland Security, and the head of any other appropriate Federal agency, determines would pose a threat similar to the threat posed by a discrete source of radium-226 to the public health and safety or the common defense and security; and

(ii) Before, on, or after August 8, 2005, is extracted or converted after extraction for use in a commercial, medical, or research activity.

Calendar Quarter(s): First quarter – January 1 through March 31

Second quarter – April 1 through June 30

Third quarter – July 1 through September 30

Fourth quarter – October 1 through December 31

Calendar Year: From January 1 through December 31.

Calibrate: To adjust and/or determine:

(1) The response or reading of an instrument relative to a series of conventionally true values; or

(2) The strength of a radiation source relative to a standard or conventionally true value.

Committed Dose Equivalent (CDE) ($H_{T,50}$): Means the dose equivalent to organs or tissues of reference (T) that will be received from intake of radioactive material by an individual during the 50 year period following the intake.

Committed Effective Dose Equivalent (CEDE) ($H_{E,50}$): The sum of the products of the weighting factors applicable to each of the body organs or tissues that are irradiated and the committed dose equivalent to these organs or tissues ($H_{E,50} = \Sigma_T W_{T,50}$).

Contact Dose Rate: A radiation dose rate as measured with the detector or instrument case within 1/2 inch of the surface being measured.

Contamination, Radioactive: Deposition of radioactive material in any place where it is not desired. Radioactive contamination may be removable (loose) or fixed.

Contaminated Area: Any area that has radioactive contamination at levels greater than the radioactivity release limits for unrestricted use.

Continuous Air Sampling/Monitoring: A method of sampling used to measure airborne radioactivity levels in routinely occupied areas.

Controlled Area: An area outside of a Restricted Area but inside the Site boundary, where access can be limited by the Licensee for any reason.

Corrective Action(s): Action(s) taken to improve areas of performance or to eliminate causes of adverse trends in performance identified during Audits, Surveillances, and as a response to a Notice of Deficiency.

Counts Per Minute (cpm): The rate of ionizing event occurrence in one minute recorded by a radiation detection instrument designed to count ionizing events caused by radiation.

Curie (Ci): A measure of the amount of radioactive material present.

One curie equals 37 billion (3.7 E+10 or 3.7×10^{10}) becquerels (dps)or 2.2 trillion (2.2 E+12) radioactive disintegration's per minute (dpm).
A millicurie (mCi) is 2.2 billion (2.2 E+09) dpm
A microcurie (μCi) is 2.2 million (2.2 E+06) dpm
A nanocurie (nCi) is 2.2 thousand (2.2 E+03) dpm
A picocurie (pCi) is 2.2 dpm.

Declared Pregnant Woman: A woman who has voluntarily informed the Licensee, in writing, of her pregnancy and the estimated date of conception. The declaration remains in effect until the declared pregnant woman withdraws the declaration in writing or is no longer pregnant.

Decontamination: Means the process of removing or reducing the level of contamination on an item or individual.

Deep Dose Equivalent (H_d): The dose equivalent at a tissue depth of 1 cm (1000 mg/cm^2) Applies to external whole body exposure.

Derived Air Concentration (DAC): The concentration of a given radionuclide in air which, if breathed by the reference man for a working year of 2,000 hours under conditions of light work (inhalation rate 1.2 cubic meters of air per hour), results in an intake of one ALI. DAC values are given in Table 1, Column 3, of appendix B to 10 CFR 20.1001-2401.

Derived Air Concentration-hour (DAC-hour): The product of the concentration of radioactive material in air (expressed as a fraction or multiple of the derived air concentration for each radionuclide) and the time of exposure to that radionuclide, in hours. The Licensee may take 2,000 DAC-hours to represent one ALI, equivalent to a committed effective dose equivalent of 5 rems (0.05 Sv).

Detector: That portion of an instrument system sensitive to and used for the quantification of ionizing radiation.

Direct Contamination Survey: This method measures fixed and removable levels of surface contamination. A direct frisk is performed by scanning the survey location using a count rate meter.

Direct Reading Dosimeter (DRD): A monitoring device consisting of a collection chamber coupled with an optical lens and calibrated scale. DRD's can be used as a device to provide individuals with an immediate estimate of their external gamma radiation exposure.

Discrete Source: A radionuclide that has been processed so that its concentration within a material has been purposely increased for use for commercial, medical, or research activities.

Disintegrations Per Minute (dpm): Refers to the number of nuclear transformations occurring per minute.

Disintegrations Per Second (dps): Refers to the number of nuclear transformations occurring per second.

Dose or Radiation Dose: A generic term that means absorbed dose, dose equivalent, effective dose equivalent, committed dose equivalent, committed effective dose equivalent, or total effective dose equivalent, as applicable to context and as defined in 10 CFR 20. The unit for absorbed dose is the rad. 100 rad = 1 Gy.

Dose Equivalent (H_T): Means the product of the absorbed dose in tissue, quality factor, and all other necessary modifying factors at the location of interest. The units for dose equivalent rem. 100 rem = 1 Sv.

Dose Rate: The quantity of absorbed dose delivered per unit of time.

Dosimeter: Any of several types of devices used to measure radiation dose. Common types include TLD, OSL, film, and direct reading dosimeters.

Effective Dose Equivalent (H_E): The sum of the products of the dose equivalent to the organ or tissue (H_T) and the weighing factors (W_T) applicable to each of the body organs or tissues that are irradiated ($H_E = \Sigma W_T H_T$).

Effluent: Material discharged into the environment from licensed operations.

Embryo/Fetus: The developing human organism from conception until the time of birth.

Exposure: Means being exposed to ionizing radiation or to radioactive material. The unit of exposure is the roentgen.

External Dose: That portion of the dose equivalent received from a source of radiation outside the body.

Extremity: Means hand, elbow, arm below the elbow, foot, knee, or leg below the knee.

Frisk: The performance of a direct survey for radioactive contamination.

Gamma Ray (Gamma Radiation): High-energy, short wavelength electromagnetic radiation (a packet of energy) emitted from the nucleus. Gamma radiation frequently accompanies alpha and beta emissions and always accompanies nuclear fission. Gamma rays are very penetrating and are best stopped or shielded against by dense materials, such as lead or uranium. Gamma rays are similar to x-rays but are usually more energetic.

General Area Dose Rate: A radiation dose rate measured at 30 cm or more from a surface.

Gray (Gy): The SI unit for absorbed dose: 1 Gy = 1 Joule kg^{-1} = 100 rad.

Groundwater: Water contained in pores or fractures in either the unsaturated or saturated zones below ground level.

In-Vitro Bioassay (indirect): The estimation of radioactivity in the human body based upon:

(1) the measurement of radioactivity in excreta or other materials taken from the body, and

(2) a biological model for the radionuclide movement in body tissues and organs.

In-Vivo Bioassay (direct): The measurement of radioactivity in the human body using instrumentation which detects radiation emitted from radionuclides in the body.

Individual Monitoring: The assessment of dose equivalent by use of devices designed to be worn by an individual; the assessment of committed effective dose equivalent by bioassay or by

determination of the time-weighted air concentrations to which an individual has been exposed; or the assessment of dose equivalent by the use of survey data.

Individual Monitoring Devices: Devices designed to be worn by a single individual for the assessment of dose equivalent. Examples include thermoluminescence dosimeters (TLD's), optically stimulated luminescent (OSL) dosimeters, direct reading dosimeters, and lapel air samplers.

Instrument: A complete system designed to quantify one or more characteristics of ionizing radiation or radioactive material.

Intake: The amount of radioactive material taken into the body by inhalation, absorption through the skin, injection, ingestion, or through wounds.

Internal Dose: That portion of the dose equivalent received from radioactive material taken into the body.

Isotopes: Nuclides having the same number of protons in their nuclei but differing in the number of neutrons. Isotopes have the same atomic number and different mass numbers.

Lens Dose Equivalent (LDE): Dose equivalent due to external exposure to the lens of the eye. It is taken as the dose equivalent at a tissue depth of 0.3 cm (300 mg/cm^2).

Licensed Radioactive Material: Source material, special nuclear material, or byproduct material received, possessed, used, transferred or disposed of under a general or specific license issued by the NRC.

License: Means the radioactive materials license issued by the NRC to the Trust to possess and/or use radioactive materials. Other licenses may be issued to the Trust by other state or federal agencies.

Licensee: The holder of the radioactive materials license (the Trust).

Limits (dose limits): The permissible upper bounds of radiation doses.

Low-Level Radioactive Waste (LLRW): Those low-level radioactive wastes containing source, special nuclear, or by-product material that are acceptable for disposal in a land disposal facility. Low-level waste has the same meaning as in the Low-Level Waste Policy Act; that is, radioactive waste not classified as high-level radioactive waste, transuranic waste, spent nuclear fuel, or by-product material as defined in paragraphs (2), (3), and (4) of the definition of *Byproduct material* set forth in 10 CFR 20.1003.

Member of the Public: An individual who is not receiving an occupational dose.

Micro: A prefix meaning "one millionth" (1 E-06), as in microcurie.

Milli: A prefix meaning "one thousandth" (1 E-03), as in millirem, millirad, or millicurie.

Minimum Detectable Activity: The smallest concentration of radioactivity in a sample that can be detected with a 5% probability of erroneously detecting radioactivity, when in fact none may be present (Type I error) and also, a 5% probability of not detecting radioactivity, when in fact it is present (Type II error). Often used interchangeably with Minimum Detectable Concentration since the difference between the two terms is only one of unit conversion.

Minor: An individual less than 18 years of age.

Monitoring (Radiation Monitoring): The measurement of radiation levels, concentrations, surface area concentrations, or quantities of radioactive material, and the use of the results of these measurements to evaluate potential exposures and doses.

Nano: A prefix meaning "one billionth" (1 E-09), as in nanocurie.

NRC: U.S. Nuclear Regulatory Commission or its duly appointed representatives.

Nuclide: Any one of the approximately 1800 isotopes of all the elements, whether radioactive or not. See radionuclide and isotope.

Occupational Dose: The dose received by an individual in the course of employment in which the individual's assigned duties involve exposure to radiation or to radioactive material from licensed and unlicensed sources of radiation, whether in the possession of the Licensee or other person. Occupational dose does not include doses received from background radiation, from any medical administration the individual has received from exposure to individuals administered radioactive material and released under 10 CFR 35.75, from voluntary participation in medical research programs, or as a member of the public.

Occupational Dose Limit: The maximum legally allowable dose to individuals during a specific time period, as defined by 10 CFR 20.

Particulate: Sometimes used to describe alpha and beta radiations, but most often used to mean dust or droplets containing radioactive material.

Pico: A prefix meaning "one trillionth" (1 E-12), as in picocurie.

Planned Special Exposure: An infrequent exposure to radiation, separate from and in addition to the annual dose limits.

Posting: A standardized sign or label which bears the standard trefoil radiation symbol in magenta or purple or black on a yellow background and information concerning a specific radiological hazard.

Protective Clothing: Clothing provided to reduce exposure and prevent the spread of contamination to personnel clothing or the body while performing work with radioactive materials.

Qualification: Certification of the fact that an individual possesses the knowledge, capabilities (e.g., physical) characteristics, or abilities gained through experience, training, or on-the-job training that an individual can perform a required task.

Qualified Escort: An individual that meets the Qualified Escort training requirements set forth in section 2.3.3.

Qualified Individual: An individual who has completed the training and/or testing requirements set forth by procedures or regulations, which in turn grants that individual permission to operate specific equipment or instrumentation, or to perform specific work duties.

Rad: The special unit of radiation dose. One rad is equal to an absorbed dose of 100 ergs/gram or 0.01 joule/kilogram (0.01 gray).

Radiation (Ionizing Radiation): Alpha particles, beta particles, gamma rays, x-rays, neutrons, high-speed electrons, high-speed protons, and other particles capable of producing ions. Radiation, as used within the context of the Radiation Protection Program does not include non-ionizing radiation such as radio or microwaves and visible, infrared, or ultraviolet light.

Radiation Area: Defined as any accessible area where the dose equivalent to an individual could exceed 5 millirem (.05 mSv) in any one hour at 30 cm from the radiation source or surface that radiation penetrates.

Radiation Safety Officer (RSO): The individual responsible for development and oversight of radiation protection program policies at the Cimarron Site. This individual shall meet the requirements set forth in NUREG-1757, Section 17.2.3.1.

Radiation Worker: An individual who has access to the Restricted Areas to perform work and has completed the training requirements listed in section 2.3.3.

Radioactive Material (49 CFR 173.403): For purposes of transportation, any material containing radionuclides where both the activity concentration and the total activity in the consignment exceed the values specified in the table in 49 CFR 173.436 or values derived according to the instructions in 49 CFR 173.433.

Radioactive Material Area: Any area or room which is posted and is used to store or contains for use an amount of licensed material exceeding 10 times the quantity of such material as listed in Appendix C to 10 CFR 20.

Radioactivity: Rate of disintegration (transformation) or decay of radioactive material. The units of activity are the curie (Ci) and the becquerel (Bq). Bq = 1 (dps) disintegration per second; Ci = 3.7×10^{10} dps.

Radionuclide: Any one of the radioactive nuclides.

Record: A document that provides evidence of the quality of services performed, demonstrates that actions were performed in accordance with radiation protection procedures, or demonstrates conformance of actions to regulatory requirements.

Reference Man: A hypothetical aggregation of human physical and physiological characteristics arrived at by international consensus. These characteristics may be used by researchers and public health workers to standardize results of experiments and to relate biological insult to a common base.

Rem: The special unit for any of the quantities expressed as dose equivalent. The dose equivalent in rems is equal to the absorbed dose in rads multiplied by the quality factor (1 rem = 0.01 sievert).

Removable Contamination Survey: The method used to measure removable contamination. Removable survey techniques are:

(1) Smear Surveys - A smear is obtained by using an absorbent filter disk to wipe with moderate pressure across the area or item to be evaluated. A smear is usually wiped over an area of 100 cm^2.

(2) Wipe Surveys – A wipe is obtained by wiping an absorbent pad or towel over a large area or the entire surface of the item being surveyed.

Respirator: An apparatus used to reduce the individual's intake of airborne radioactive materials.

Restricted Area: An area having access controlled by the Licensee for the purpose of protecting individuals against undue risk from exposure to radiation and radioactive materials. Restricted Area does not include areas used as residential quarters, but separate rooms in a residential building may be set apart as a Restricted Area.

Sealed Source: Any by-product material that is encased in a capsule designed to prevent leakage or escape of the by-product material.

Shallow Dose Equivalent (SDE): The dose equivalent at a tissue depth of 0.007cm (7 mg/cm^2), averaged over an area of one square centimeter. It applies to external exposure of the skin of the whole body or of an extremity.

Sievert (Sv): The SI unit of any of the quantities expressed as dose equivalent. The dose equivalent in sieverts is equal to the absorbed dose in grays multiplied by the quality factor. 1 Sv = 100 rem.

Site Boundary: The line beyond which the land or property is not owned, leased, or otherwise controlled by the Licensee.

Skin of the Whole Body: The skin of the whole body, exclusive of skin of the extremities.

Smear: A radiation survey technique which is used to determine levels of removable surface contamination. A medium (typically filter paper) is rubbed over a surface (typically of area 100 cm^2), followed by a quantification of the activity on the medium. Also known as a swipe.

Source Material:

(1) Uranium or thorium or any combination of uranium and thorium in any physical or chemical form; or

(2) Ores that contain, by weight, one-twentieth of 1 percent (0.05 percent), or more, of uranium, thorium, or any combination of uranium and thorium. Source material does not include special nuclear material.

Special Nuclear Material:

(1) Plutonium, uranium-233, uranium enriched in the isotope 233 or in the isotope 235, and any other material that the Commission, pursuant to the provisions of section 51 of the Act, determines to be special nuclear material, but does not include source material; or

(2) Any material artificially enriched by any of the foregoing but does not include source material.

Stochastic Effects: Health effects that occur randomly and for which the probability of the effect occurring, rather than its severity, is assumed to be a linear function of dose without threshold. Hereditary effects and cancer are examples of stochastic effects.

Survey: An evaluation of the radiological conditions and potential hazards incident to the production, use, transfer, release, disposal, or presence of radioactive materials or other sources of radiation. When appropriate, such an evaluation includes a physical survey of the location of a source of radiation and measurements or calculations of levels of radiation or concentrations or quantities of radioactive material present.

Thermoluminescent Dosimeter (TLD): An integrating detector where radiation energy is absorbed (trapped) and can be read out later by thermal excitation of the detector.

Total Effective Dose Equivalent (TEDE): The sum of the deep dose equivalent (for external exposures) and the committed effective dose equivalent (for internal exposures).

Total Organ Dose Equivalent (TODE): The sum of the deep dose equivalent and the committed dose equivalent to the organ receiving the highest dose.

Unrestricted Area: Any area to which access is not limited or controlled for purposes of protection of individuals from exposure to radiation and radioactive materials.

Uptake: Quantity of a radionuclide taken up by the systematic circulation (e.g., by injection into the blood, by absorption from compartments in the respiratory or gastrointestinal tracts, or by absorption through the skin or through wounds in the skin).

Uranium (Natural, Depleted and Enriched):

Natural Uranium: Uranium found in nature. Natural uranium contains 0.71 weight percent U-235, 99.3 weight percent U-238, and a trace of U-234.

Depleted Uranium: Uranium in which the U-235 isotope represents less than 0.71 weight percent of the mass of the material. Depleted uranium is less radioactive than natural uranium.

Enriched Uranium: Uranium in which the U-235 isotope represents greater than 0.71 weight percent of the mass of the material. The alpha emission rate increases from 1.5 E3 dpm per mg at 0.71 weight percent enrichment to 1.4 E5 dpm per mg at 93% enrichment.

Visitor: An individual who is not an employee or contractor of the Licensee.

Week: Seven consecutive days starting on Sunday.

Weighting Factor (W_T): The proportion of risk of stochastic effects resulting from irradiation of the organ or tissue to the total risk of stochastic effects when the whole body is irradiated uniformly.

Whole Body (WB): Means, for purposes of whole body exposure, the head, trunk (including male gonads), arms above the elbow, or legs above the knee.

Year: The period of time beginning on January 1 and ending on December 31 that is used to determine compliance with the NRC.

X-Ray: Penetrating electromagnetic radiation having a wavelength much shorter than that of visible light. X-rays are usually produced by an excitation of the electron field around certain nuclei. In nuclear reactions, it is customary to refer to photons originating in the electron field of the atom as X-rays.

17.0 REFERENCES

1. 10 CFR 19, "Notices, Instructions and Reports to Workers; Inspection and Investigations"
2. 10 CFR 20, "Standards for Protection Against Radiation"
3. 10 CFR 70, "Domestic Licensing of Special Nuclear Material"
4. "Cimarron Facility Environmental Response Trust Decommissioning Plan," Revision 3.
5. EPM017-CALC-001, "Dose Rate Near Uranium Treatment Train"
6. EPM028-CALC-001, "Potential Intake Calculation"
7. NCRP 87-1987, "Use of Bioassay Procedures for Assessment of Internal Radionuclide Deposition"
8. NUREG/CR-0041, Rev. 1, "Manual of Respiratory Protection Against Airborne Radioactive Material"
9. NUREG-1400, "Air Sampling in the Workplace"
10. NUREG 1757, "Decommissioning Process for Materials Licensees"
11. NUREG-1507, Rev. 1, "Minimum Detectable Concentrations with Typical Radiation Survey Instruments for Various Contaminants and Field Conditions," August 2020
12. NUREG-1556, Vol. 7, "Consolidated Guidance About Materials Licenses, Program-Specific Guidance About Academic, Research and Development, and Other Licenses of Limited Scope Including Gas Chromatographs and X-Ray Fluorescence Analyzers," Appendix H, "Sample Audit Program," 2018
13. Regulatory Guide 4.15, Rev. 2, "Quality Assurance for Radiological Monitoring Programs (Normal Operations) - Effluent Streams and the Environment"

14. Regulatory Guide 8.15, Rev. 1, "Acceptable Programs for Respiratory Protection"
15. Regulatory Guide 8.25, Rev. 1, "Air Sampling in the Workplace"
16. Regulatory Guide 8.28, Rev. 0, "Audible Alarm Dosimeters"
17. Regulatory Guide 8.34, Rev. 0, "Monitoring Criteria and Methods to Calculate Occupational Radiation Doses"Regulatory Guide 8.36, Rev. 0, "Radiation Dose to the Embryo/Fetus"
18. The Cimarron Environmental Response Trust Special Nuclear Material License (SNM-928)
19. U.S. NRC, "Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of License for Byproduct, Source or Special Nuclear Material," August 1987
20. U.S. NRC, "Disposal or Onsite Storage of Thorium or Uranium Wastes from Past Operations," October 1981
21. U.S. EPA, "Federal Guidance Report No. 11," 1988

APPENDIX A

POTENTIAL INTAKE CALCULATION (EPM028-CALC-001)

ENERCON Excellence—Every project. Every day.	CALCULATION COVER SHEET	CALC NO.	EPM028-CALC-001
		REV.	4
		PAGE NO.	1 of 14

Title:	Potential Intake Calculation	Client:	Cimarron Environmental Response Trust
		Project Identifier:	EPM028

Item	Cover Sheet Items	Yes	No
1	Does this calculation contain any open assumptions, including preliminary information, that require confirmation? (If **YES**, identify the assumptions.)	☐	☒
2	Does this calculation serve as an "Alternate Calculation"? (If **YES**, identify the design verified calculation.) **Design Verified Calculation No.** _____	☐	☒
3	Does this calculation supersede an existing Calculation? (If **YES**, identify the design verified calculation.) **Superseded Calculation No.** EPM028-CALC-001, Rev. 3	☒	☐

Scope of Revision:

The calculation has been updated to reflect a dispersitblity factor change from 0.1 to 1 in response to NRC comments and address more current information from DP, Rev. 3 and RPP, Rev. 5.

Revision Impact on Results:

Conclusions related to chemical intakes of uranium do not change. Potential intakes of Tc-99 and uranium daughters are significantly less than 1% ALI.

Study Calculation	☐	**Final Calculation**	☒
Safety-Related	☐	**Non-Safety-Related**	☒

(Print Name and Sign)

Originator: Jay Maisler	**Date:**
Design Reviewer: A. Joseph Nardi	**Date:**
Approver: Dane Watson	**Date:**

ENERCON Excellence—Every project, Every day.	CALCULATION REVISION STATUS SHEET	CALC NO. EPM028-CALC-001
		REV. 4
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CALCULATION REVISION STATUS

REVISION	DATE	DESCRIPTION
0	3/26/2019	Initial Issue
1	6/20/2019	Corrected mass intake from inhalation and made editorial corrections
2	11/22/2019	Added calculation of potential intake for Tc-99 and sensitivity analysis for potential intakes of U-235 and U-238 progeny with corresponding Attachments.
3	5/6/2021	Corrected page numbering on throughout and in corrected Table of Contents. Corrected headers in Attachments. Corrected CALC number on Calculation Preparation Checklist.
4	4/8/2022	Updated assumptions and input for consistency with D-Plan, Rev. 3 and RPP, Rev. 5. Corrected editorial errors and omissions. Changed dispersibilty factor (D) from 0.1 to 1 in response to NRC staff comments. Attachments A, B and C are replaced in their entirety. Added Attachment D with specifications for resin.

PAGE REVISION STATUS

PAGE NO.	REVISION	PAGE NO.	REVISION
1-14	4		

APPENDIX/ATTACHMENT REVISION STATUS

APPENDIX NO.	NO. OF PAGES	REVISION NO.	ATTACHMENT NO.	NO. OF PAGES	REVISION NO.
			Attachment A	2	4
			Attachment B	1	4
			Attachment C	2	4
			Attachment D	3	4

CALCULATION TABLE OF CONTENTS

1.0 Purpose and Scope

The purpose of this calculation is to estimate potential intakes of uranium from groundwater processing and resin handling at the Cimarron Site. This calculation also provides the potential intake from handling spent resin potentially contaminated with Tc-99. The methodology for potential intake and need for air sampling is based on the methodology provided in NUREG-1400 (Ref. 3.1).

A sensitivity analysis was performed to determine the contribution to potential intakes from airborne exposure to thorium and protactinium progeny of U-235 and U-238.

2.0 Summary of Results and Conclusions

The potential intake calculation is based on a conservative scenario that assumes that each spent resin vessel is loaded to the the site limit of 1,200 grams of U-235 (as specified in NRC License No. SNM-928, Amendment 21) (Ref. 3.2), the potential intake from inhalation of 5% enriched uranium is 0.11% of the corresponding annual limit on intake (ALI) (Ref. 3.3). A more realistic potential intake calculation based on a maximum of 500 grams of U-235 (the maximum limit as started in the D-Plan), potential intake from inhalation of 2.9% enriched uranium is 0.05% ALI.

The sensitivity analysis based on the conservative assumption (1,200 grams U-235; 5% enriched uranium) determined that the potential intake contribution from short-lived progeny of U-235 and U-238 decay products is insignificant (very small fractions of an ALI) in comparison to the potential intake from uranium inhalation.

The potential intake from inhalation of Tc-99 is insignificant (e.g., very small fractions of an ALI) (Ref. 3.3).

Analysis of potential exposure through ingestion of contaminated groundwater could pose a concern related to the limit of 10 mg per week for uranium. The potential intake from drinking 2.84 liters of water in a week was conservatively calculated to meet the weekly limit. For the more realistic calculation, drinking 9.82 liters in a week would result in a potential intake of 10 mg of uranium. Consumption of groundwater is not permitted at the Site.

For Tc-99, daily water consumption would be at volumes that are not possible even if consumption was permitted. An individual would need to drink 854 liters of contaminated water per day for 365 days to receive a potential intake of 2% ALI.

3.0 References

3.1 NUREG-1400, Air Sampling in the Workplace, September 1993.
3.2 U.S. Nuclear Regulatory Commission Materials License Number SNM-928, Amendment 21
3.3 10 CFR 20, Appendix B, Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage
3.4 Cimarron Radiation Protection Plan, draft Rev. 5
3.5 BA1 Isotopic Data for Enercon.xlsx, Isotopic abundance of Uranium source provided as an Excel spreadsheet
3.6 Cimarron Facility Decommissioning Plan, Rev. 3
3.7 DOE-STD-1136-2017 Guide to Good Practices for Occupational Radiological Protection in Uranium Facilities
3.8 Dow Product Data Sheet, AMBERSEP™ 21K Ion Exchange Resins, Rev. 0. January 2018
3.9 10 CFR 20.1201(e)

4.0 Assumptions

4.1 The maximum groundwater uranium mass concentration is assumed to be 3,516 µg/L. This is conservative based on the highest reported groundwater concentration found in BA1. (Ref. 3.6, section 3.5.3) and is used in the conservative potential intake calculation. The projected uranium mass concentration in BA1 is 1,018 µg/L, which is used in the realistic calculation. (Ref. 3.6, section 8.6.3).

4.2 The average groundwater Tc-99 mass concentration is assumed to be 15 ng/L (Ref. 3.6, section 6.2), which converts to 257 pCi/L.

4.3 For the conservative conservative calculation, the maximum mass of uranium in one spent resin vessel is assumed to be 1,200 g, the Site possession limit under the license (Ref. 3.2). The maximum mass of U-235 assumed to be present in the spent resin being processed for disposal is 500 grams is used in the realistic calculation. This is higher than the maximum mass estimated in the treatability study. (Ref. 3.6, section 8.3.2)

4.4 All radionuclides considered in this calculation and sensitivity analysis are assumed to be soluble in groundwater.

4.5 Determining "potential intake" for uranium, based on the NUREG-1400 methodology (Ref. 3.1), requires various assumptions. These assumptions are used in the sensitivity analysis:

4.5.1 Total activity processed is based on eight spent resin bed vessels processed per year. (Ref. 3.6, section 5.6.12)

4.5.2 "Release Fraction," R, is based on "non-volatile powder" – 0.01. The spent resin is mixed with an inert material and is handled in a confined system.

4.5.3 "Confinement Factor," C, is equivalent to a "well-vented hood" – 0.1. This is based on equipment designed to confine spent resin within the process system.

4.5.4 "Dispersibility," D, is a factor that considers dispersiblity that comes from adding energy to the system through grinding, milling, boilinig, or exothermice reactions. As no energy is provided to the system, a factor of D = 1 is applied to the calculationbased on the contaminant being moist resin – 0.1. (NOTE: A value of 1 was chosen because, as discussed in the Cimarron Facility Decommissioning Plan (Ref. 3.6, Section 8.7.2), spent resin is handled in a closed system and does not come into contact with workers Additionally, as shown in the Dow Product Data Sheet for AMBERSEP™ 21K Ion Exchange Resins (Attachment D) (Ref. 3.8), the size of the resin beads, is 575 ± 50 µm. If released from the closed system, the large resin beads will rapidly fall out of the air and are not respirable when no energy is added to this gravity-fed system. The uranium is adhered to the resin beads and will not be released into the air. This is approach is consistent with the discussion provided in Section 1.2.3 of Reference 3.1.

4.6 Determining "potential intake" for Tc-99, based on the NUREG-1400 methodology (Ref. 3.1), requires various assumptions:

4.6.1 Total activity processed is based on groundwater flow of 125 gallons per minute processed for cleanup. (Ref. 3.6, section 8.6.2)

4.6.2 Total volume of groundwater processed in a year assumes continuous processing for a year – 365 days, 24 hours per day.

4.6.3 "Release Fraction," R, is based on "liquid" – 0.01.

4.6.4 No credit for "Confinement Factor," C, is taken – 1.

4.6.5 No credit is taken for "Dispersibility," D = 1.

4.7 The ALI and DAC used for inhalation are for U-238, class Y. U-234, U-235, and U-238 have the same value for these ALIs and DACs (Ref. 3.3), therefore isotopic distribution is irrelevant to the dosimeteric calculation.

4.8 The solubility class for the Tc-99 ALI is W (Ref. 3.3), the most restrictive inhalation ALI.

4.9 Isotopic distribution for U-235 progeny and U-238 progeny in the sensitivity analysis only considers isotopes in the decay chain that are in secular equilibrium with their uranium parents; specifically, Th-231, Th-234, Pa-234m.

4.10 The ALI for Pa-234 is used as there is no ALI provided for Pa-234m in 10 CFR 20 Appendix B (Ref. 3.3).

5.0 Design Inputs

5.1 Annual Limit on Intake (ALI) for Inhalation

Consistent with Reference 3.7, Table 2-10 provides inhalation classification for some uranium compounds. All soluble uranium salts, such as are present in groundwater at the Cimarron Site, are "Type F," which is equivalent to "Class D" used in NRC regulations (Ref. 3.3). The calclulation uses Class Y for uranium and progeny and Class W for Tc-99. These are conservative (bounding) conditions. The following properties are taken from Reference 3.3.

Radionuclide	Class Y ALI
U-234, U-235, U-238	4.00 E-02 µCi
Tc-99	7.00 E+02 µCi
Th-231	6.00 E+03 µC i
Th-234	2.00 E+02 µCi
Pa-234m (Pa-234)	7.00 E+03 µCi

5.2 ALI for Oral Ingestion

The following values for oral intakes are taken from Reference 3.3:

1.00 E01 µCi (class D, bone surface) (Uranium)
2.00 E01 µCi (class D) (Uranium)
4.00 E+03 µCi (class D) (Tc-99)

6.0 Methodology

6.1 Potential Intake – Inhalation

Two calculations were performed, a conservative calculation that assumes groundwater containing uranium enriched to 5% U-235 such that 1,200 grams of U-235 (Site license possession limit) is accumulated on every spent resin bed processed during the year. A realistic calculation assumes that the uranium enrichment is 2.9% U-235 and that each spent resin bed accumulates 500 grams of U-235.

Reference 3.1 provides calculational methods to support decisions related to the need to perform air sampling at a facility and determine the potential intake for workers.

The potential intake calculation involves determining if the amount of unsealed radioactive materials handled in a year would indicate the need for performing air sampling. Following this methodology, the amount uranium on a spent resin bed was determined. Then, as discussed in Assumption 4.5.1, the total amount of uranium handled in a year was determined. Other assumptions (4.5.2, 4.5.3, 4.5.4) were used to calculate "potential intake" from inhalation.

The total activity of Tc-99 handled in a year is determined based on the average concentration of Tc-99 in groundwater provided in Assumption 4.2 multiplied by the total volume of groundwater processed during a year (Assumptions 4.6.1 and 4.6.2) Other assumptions (4.6.3, 4.6.4, 4.6.5) were used to calculate "potential intake" from inhalation.

The potential intake calculation also provides information related to the need (or abence thereof) of monitoring workers for internal dose.

6.2 Oral Intake Consumption

The amount of untreated groundwater that would need to be consumed for an individual to have an intake of 2% ALI and 1 ALI was calculated based on Assumption 4.1 for uranium and Assumption 4.2 for Tc-99.

6.3 Spent Resin Loading

The total amount of uranium that could be loaded onto a spent resin bed was calculated for 2.9% and 5% enriched uranium. These calculations provided for conservatism in the potential intake calculation performed.

The specific activity for uranium based on enrichment was determined based on the methodology provided in Reference 3.7, Example 1.

6.4 Chemical Intake

Intake of soluble uranium is limited to 10 mg per week as required by 10 CFR 20.1201(e) (Ref. 3.9). Using the "potential intake" from inhalation, the mass that an individual could intake in a year was calculated. Because the limit is based on a weekly limit, ingestion of contaminated water is more limiting than inhalation. Based on Assumption 4.1, the amount of contaminated water that would need to be consumed during a week to ingest 10 mg of soluble uranium was calculated.

6.5 Sensitivity Analysis

The sensitivity analysis considers the potential intake from inhalation of U-234, U-235 and its progeny (Th-231), and U-238 and its progeny (Th-234, Pa-234m). The progeny for U-235 and U-238 are considered to be in secular equilibrium with their respective parent. Activity distribution percentages are based on 5% enrichment, which provides an upper bound for potential intake considerations.

7.0 Calculation

7.1 Calculation Inputs

Activity concentration is determined base on enrichment. The specific activity for the enriched value is divided by the mass concentration. The specific activity for uranium based on enrichment is calculated using Equation 7.1. (Ref. 3.1, Example 1)

$$SA = 0.4 + 0.38E + 0.0034E^2$$

Equation 7.1

Where *SA* of enriched uranium is the specific activity in Ci/g for enrichment, *E* in percent. For 5% enriched uranium, *SA* = 2.39 E-06 Ci/g; for 2.9% enrichment, *SA* = 1.53 E-06 Ci/g.

For the conservative activity concentration determination, the maximum mass concentration reported at the Site is used (Assumption 4.1). For the realistic activity concentration, the projected uranium mass concentration is used (Assumption 4.1)

$$Activity\ Concentration\ \left(\frac{\mu Ci}{mL}\right) = \frac{Mass\ Concentration\ \left(\frac{pCi}{L}\right)}{SA\ \left(\frac{Ci}{g}\right)} \times 1E-03\ (unit\ conversions)$$

Equation 7.2

For the conservative calculation, a activity concentration in influent results in 8.39 E-06 µCi/mL. For the realistic calculation, the activity concentration is 1.56 E-06 µCi/mL.

7.2 Oral Intake Consumption

The oral intake consumption is found using the Equation 7.3.

$$I_o = \frac{ALI}{C_{max}} \times 1E06$$

Equation 7.3

Where I_o, in liters, is equal to annual limit on intake (ALI) µCi for U-238 or Tc-99 divided by the activity concentration (pCi/L) in groundwater multiplied by a 1E+06 pCi/µCi.

The results from the conservative calculation for uranium indicate that drinking 23.85 L of contaminated groundwater at maximum activity would result in an individual intake of 2% ALI; 1,193 L would need to be consumed to have an intake of 1 ALI. Consuming groundwater at the Site is prohibited. Volumes of water for the realistic calculation are significantly greater than the conservative calculation; 128 L for 2% ALI and 6,420 L for 1 ALI.

The results of this calculation for Tc-99 indicate that drinking 3.12 E+05 L of contaminated groundwater would result in an individual intake of 2% ALI; 1.56 E+07 L would need to be consumed to have an intake of 1 ALI. The 2% ALI scenario would require consumption of 854 L per day for 365 days; 42,700 L per day for 365 daysr for an intake of 1 ALI.

7.3 Spent Resin Loading

The mass of total uranium in a fully spent resin bed is found using Equation 7.4.

$$M = \frac{U^{235}\ Mass\ Assumption}{E}$$

Equation 7.4

Where the total mass of uranium in spent resin vessel, *M*, equals the mass of U-235 on the spent resin bed (Assumption 4.3) divided by the enrichment, *E*.

Based on 5% enrichment, the uranium mass in one spent resin bed is 24 kg; for 2.9%, the uranium mass is 17.2 kg.

7.4 NUREG-1400 Methodology

7.4.1 Potential Intake for Uranium

The total activity in spent resin processed in a year is based on eight resin bed exchangesduring the first year of operation. (Ref. 3.6, section 5.6.12) The total activity handled during a year is calculated with Equation 7.5.

$$Q = SA_{enrich} \times M \times 8$$

Equation 7.5

Where Q, the total activity handled in a year (Ci) is equal to the specific activity based on enrichment, SA_{enrich}, Ci/g multiplied by the total mass of uranium on a fully spent resin bed from Equation 7.4. The activity Q is compared to the ALI for uranium to determine if it exceeds 10^4 times the ALI. If it does, then air sampling should be considered per Reference 3.1.

The total activity assuming 5% and 2.9% enrichment exceed 10^4 times the ALI. Air sampling should be considered and is addressed in the RPP (Ref. 3.6, section 8.3.3)

Potential intake from inhalation is determined from Equation 7.6 (Ref. 3.1, Equation 1.2).

$$I_p = Q \times R \times C \times D \times 10^{-6}$$

Equation 7.6

Where I_p is the potential intake from inhalation in µCi, *Q* is the total activity from Equation 7.5 converted to µCi, *R* is the release fraction (Assumption 4.5.2), *C* is the confinement factor (Assumption 4.5.3), and *D* is dispersibility (Assumption 4.5.4). 10^{-6} is an additional factor provided in Reference. 3.1

The potential intake from inhalation of uranium for 5% enrichment was more limiting than 2.9% enrichment. The results for 5% enriched uranium are 4.58 E-05 µCi in a year, which is 0.11% ALI. At 2.9% enrichment, the potential intake was calculated to be 2.11 E-05 µCi in a year or 0.05% ALI.

7.4.2 Potential Intake for Tc-99

The total activity concentration for Tc-99 is calculated by multiplying the Tc-99 groundwater mass concentration (Assumption 4.2) the specific activity for Tc-99 (Ref. 3.6, section 6.2) converted to µCi/mL. The total activity concentration is multiplied by the groundwater flow times the number of minutes in a year (Assumption 4.6.1 times Assumption 4.6.2) to calculate total activity for Tc-99 handled in a year.

Potential intake from inhalation is determined from Equation 7.6.

Where I_p is the potential intake from inhalation in µCi, Q is the total activity as discussed converted to µCi, R is the release fraction (Assumption 4.6.3), C is the confinement factor (Assumption 4.6.4), and D is dispersibility (Assumtpion 4.6.5). 10^{-6} is an additional factor provided in Reference 3.1

The potential intake of Tc-99 is 6.38 E-04 µCi in a year, which is an insignificant fraction of an ALI (9.11 E-05%).

7.5 Chemical Intake

Chemical intake from inhalation is calculated by dividing the potential intake by specific activity to determine the mass of uranium inhaled in a year. The results demonstrate that significantly less than 10 mg is potentially inhaled for an entire year; 1.92 E-05 g for the conservative calculation; 1.38 E-05 g for the realistic calculation.

The amount of groundwater that would need to be consumed to have an intake of 10 mg of soluble uranium was determined using Equation 7.7.

$$V = \frac{10\ mg}{M_c\ \mu g/L} \times 10^3$$

Equation 7.7

Where V is the volume consumed in liters. 10 mg is the weekly soluble uranium intake limit (10 CFR 20.1201(e) (Ref. 3.9)). M_c is the mass concentration of uranium in groundwater at BA1 (Assumption 4.1).

The conservative calculation resulted in a weekly intake exceeding 10 mg soluble uranium if an individual consumed 2.84 L of contaminated water. The realistic calculation resulted in a weekly intake exceeding the limit if an individual consumed 9.82 L of contaminated water. Consumption of groundwater at the Cimarron site is prohibited.

7.6 Sensitivity Analysis

The sensitivity analysis was performed assuming 5% enriched uranium and the following::

- Th-231 is in secular equilibrium with U-235 (i.e., the activity of this progeny is equal to the activity of U-235).
- Th-234 and Pa-234m are in secular equilibrium with U-238 (i.e., the activity of each of these progeny is equal to the activity of U-238).

Total uranium activity was determined for the potential intake calculation. Isotopic activity was determined by calculating the activity fraction for each of the uranium isotopes (U-234, U-235, and U-238) and multiplying by the total uranium activity.

Activity fractions were calculated using the equations provided in Example 3a of Reference 3.7:

$$AF_{234} = \frac{Enrichment_{234} * SA_{234}}{Enrichment_{234} * SA_{234} + Enrichment_{235} * SA_{235} + Enrichment_{238} * SA_{238}}$$

Equation 7.8

$$AF_{235} = \frac{Enrichment_{235} * SA_{235}}{Enrichment_{234} * SA_{234} + Enrichment_{235} * SA_{235} + Enrichment_{238} * SA_{238}}$$

Equation 7.9

$$AF_{238} = \frac{Enrichment_{238} * SA_{238}}{Enrichment_{234} * SA_{234} + Enrichment_{235} * SA_{235} + Enrichment_{238} * SA_{238}}$$

Equation 7.10

Where AF_{234} is the activity fraction for U-234; AF_{235} is the activity fraction for U-235; AF_{238} is the activity fraction for U-238. Enrichment is by percent weight. SA_{234} is the specific activity of U-234 (2.30 E+08 Bq/g); SA_{235} is the specific activity of U-235 (79,312 Bq/g); SA_{238} is the specific activity of U-238 (12,329 Bq/g). (Ref. 3.7)

Enrichment for U-234, based on 5% U-235 enrichment, is calculated by the following equation provided in Example 3a of Reference 3.7:

$$Enrichment_{234} = \frac{0.0055\_natural\_fraction\_^{234}U}{0.72\_natural\_fraction\_^{235}U} * 5.0 * 1.2$$

Equation 7.11

$$Enrichment_{238} = 100 - Enrichment_{234} - 5.0$$

Equation 7.12

The NUREG-1400 methodology (Ref. 3.1) discussed in Section 7.4 was applied for each of the uranium istopes and the U-235 and U-238 progeny. The results of the analysis for each of the isotopes is provided below, as a percentage of ALI:

- U-234: 0.10% ALI
- U-235: <0.01% ALI
- Th-231: <0.01% ALI
- U-238: 0.01% ALI
- Th-234: <0.01% ALI
- Pa-234m: <0.01% ALI

The fractional contribution to the ALI for each of the radionuclides, shown as <0.01%, above do not change the total rounded result of 0.11% ALI. The sum of fractions for the ALIs from all uranium isotopes and progeny is 1.14 E-03 which is the same value as for only the uranium isotopes.

8.0 Computer Software

A Microsoft Excel spreadsheet was used to perform calculations and sensitivity discussed in this calculation. The individual spreadsheets are provided in Attachments A, B, and C.

ENERCON Excellence—Every project. Every day.	**ATTACHMENT A** Potential Intake Calculation Uranium	**CALC NO.** EPM028-CALC-001 **REV.** 4 **PAGE NO.** 1 of 2

Calculation Inputs	Conservative calculation (5%)				More realistic calculation (2.9%)				
Specific Activity (5% enrich.)	2.39E-06	Ci/g			SA (2.9%)	1.53E-06	Ci/g		
Max mass conc. - BA1	3516	µg/L				1018	µg/L		
Activity concentration	8.39E-06	µCi/mL				1.56E-06	µCi/mL		
Oral Intake ALI	1.E+01	µCi				1.E+01	µCi		
Consumption for 2% ALI	23.85	L				128.36	L		
Consumption for 1 ALI	1192.51	L				6.42E+03	L		
Spent resin loading									
U-235 mass assumption	1200	g			Max U235	500	g		
Total U assume 5%	2.40E+04	g			U (2.9%)	1.72E+04	g		
NUREG-1400 Methodology									
Total activity per resin bed	5.72E-02	Ci	5.72E+04	µCi		2.64E-02	Ci	2.64E+04	µCi
Total activity (Q) (8 bed/y)	4.58E-01	Ci	4.58E+05	µCi		2.11E-01	Ci	2.11E+05	µCi
Is Q > 1E+04 ALI?	1.14E+07	YES				5.28E+06	YES		
Potential Intake (I_P)	I_P = Q x 1E-06 x R x C x D								
Total activity (Q) (8 bed/y)	4.58E+05	µCi				2.11E+05	µCi		
Release Fraction (R)	0.01	Nonvolatile Powder				0.01	Nonvolatile Powder		
Confinement Factor (C)	0.1	Well-ventilated hood	Glovebox would be 0.01			0.1	Well-ventilated hood		
Dispersibility (D)	1	No energy added to system.				1	No energy added to system.		
Potential Intake (I_P)	4.58E-04	µCi				2.11E-04	µCi		
%ALI	1.14%	1.14E-02	Fraction			0.53%	5.28E-03	Fraction	

Chemical Intake	Conservative calculation (5%)					More realistic calculation (2.9%)			
Potential Intake (I_P)	4.58E-04	µCi				2.11E-04	µCi		
Mass intake per year	1.92E-04	g	1.92E-01	mg		1.38E-04	g	1.38E-01	mg
Drinking	3516	µg/L				1018	µg/L		
Limit per week	10	mg				10	mg		
Volume per week	2.84E+00	L				9.82E+00	L		
Activity/week	2.39E-08	Ci				1.53E-08	Ci		
	2.39E-02	µCi				1.53E-02	µCi		
%ALI	0.24%					0.15%			

Assumptions				
Mass concentration	15	**ng/L**		
Specific Actvity	1.71E+10	**pCi/g**		
Activity concentration	2.57E+02	**pCi/L**		
Groundwater flow	125	**gpm**		
Groundwater vol/year	6.57E+07	**gallons**	2.49E+08	**L**
Conversion	3.785	**L/g**		
NUREG-1400 Methodology				
Tc-99 Activty/y	6.38E+10	**pCi**	6.38E+04	**µCi**
ALI	7.00E+02	**µCi**		
Is Q > 1E+04 ALI?	9.11E+01	**NO**		
Potential Intake (I_P)	I_P = Q x 1E-06 x R x C x D			
Release Fraction (R)	0.01	Liquid		
Confinement Factor (C)	1	No credit taken		
Dispersibility (D)	1	No energy added to system.		
Potential Intake (I_P)	6.38E-04	**µCi**		
%ALI	0.00%	9.11E-07	Fraction	
Oral Intake				
Activity concentration	2.57E+02	pCi/L		
Oral ALI	4.00E+03	**µCi**		
Consumption for 2% ALI	3.12E+05	**L**		
Consumption for 1 ALI	1.56E+07	**L**		
Consumption per day (2%)	8.54E+02	**L/day for 365 days/y**		
Consumption per day (ALI)	4.27E+04	**L/day for 365 days/y**		

Specific Activity for Enriched Uranium (DOE-STD-1136-2017, Example 1, p. 9)	
$SA=[0.4 + 0.38E + 0.0034E^2] \times 10^{-6}$ Ci/g (E is enrichment percent)	
Enrichment	SA (Ci/g)
2.9	1.53E-06
5	2.39E-06

Isotope	Spec. Activity	SA Units	Mass Fraction	Activity Fraction
U-234	2.30E+08	Bq/g	0.05%	87.06%
U-235	79312	Bq/g	5.00%	3.27%
U-238	12329	Bq/g	94.95%	9.67%

Activity Fraction Determination (Source: DOE-STD-1136-2017, Ex. 3a, p. 28)

Sensitivity Analysis - 5% enrichment
Assumptions

5.72E-02	Ci	Total U-activity in one spent resin bed			
87.1%	U-234	Activity % of U-234 in Uranium			
3.3%	U-235	Activity % of U-235 in Uranium			
9.7%	U-238	Activity % of U-238 in Uranium			
4.98E-02	Ci	U-234	ALI(Y)	4.E-02	µCi
1.87E-03	Ci	U-235	ALI(Y)	4.E-02	µCi
1.87E-03	Ci	Th-231	ALI(W,Y)	6.E+03	µCi
5.53E-03	Ci	U-238	ALI(Y)	4.E-02	µCi
5.53E-03	Ci	Th-234	ALI(W,Y)	2.E+02	µCi
5.53E-03	Ci	Pa-234m	ALI(Y)	7.E+03	µCi

		U-234	U-235	Th-231	U-238	Th-234	Pa-234m
Total activity per resin bed	**µCi**	4.98E+04	1.87E+03	1.87E+03	5.53E+03	5.53E+03	5.53E+03
Total activity (Q) (8bed/y)	**µCi**	3.99E+05	1.50E+04	1.50E+04	4.43E+04	4.43E+04	4.43E+04
	Q/ALI	9.97E+06	3.75E+05	2.50E+00	1.11E+06	2.21E+02	6.32E+00
	Is Q > 1E+04 ALI?	**YES**	**YES**	**NO**	**YES**	**NO**	**NO**
Potential Intake (I_P= Q x 1E-06 x R x C x D)							
Total activity (Q) (8bed/y)	**µCi**	3.99E+05	1.50E+04	1.50E+04	4.43E+04	4.43E+04	4.43E+04
Release Fraction (R)	Nonvolatile Powder	0.01	0.01	0.01	0.01	0.01	0.01
Confinement Factor (C)	Well-ventilated hood	0.1	0.1	0.1	0.1	0.1	0.1
Dispersibility (D)	No energy added	1	1	1	1	1	1
Potential Intake (I_P)	**µCi**	3.99E-04	1.50E-05	1.50E-05	4.43E-05	4.43E-05	4.43E-05
	Fraction of ALI	9.97E-03	3.75E-4	2.50E-09	1.11E-03	2.21E-07	6.32E-09
	Percent ALI	1.00%	0.04%	0.00%	0.11%	0.00%	0.00%
	All U-isotopes (µCi)	4.58E-04	All isotopes	5.61E-04	Sums Potential Intake for all radioisotopes		
	Fraction of ALI	1.14E-02	Fraction of ALI	1.14E-02	Sums Fraction of ALI for all radioisotopes		
	Percent ALI	1.14%	Percent ALI	1.14%	Sums Percent ALI for all radioisotopes		

Product Data Sheet

AMBERSEP™ 21K Ion Exchange Resins

Industrial-grade, Strong Base Anion Exchange Resins for Mineral Processing Applications

Description

AMBERSEP™ 21K Ion Exchange Resins are Type I strong base anion resins with excellent kinetics and regeneration efficiency, along with outstanding physical stability. Both are especially suited for mineral processing and groundwater remediation applications due to their enhanced-porosity gel bead matrix made by a special process giving fast equilibrium rates and improved resistance to organics.

AMBERSEP™ 21K 16–20 Ion Exchange Resin, with its screened particle size from 16 – 20 U.S. Mesh, is a high-efficiency, large-bead resin suitable for fluidized-bed and Resin-In-Pulp (RIP) applications.

AMBERSEP™ 21K XLT Ion Exchange Resin, with its high capacity and uniform particle size, represents the state-of-the-art solution for mineral processing, giving enhanced performance for packed bed systems.

Applications

- Mineral Processing (Zn, Mn, etc.)
- Precious metal recovery (Au, Ag, Pt, Pd, Rh)
- Uranium recovery

Typical Physical and Chemical Properties**

Matrix	Styrene-divinylbenzene, gel
Type	Strong base anion, Type I
Functional Group	Quaternary amine
Physical Form	Opaque, white to tan, hard, spherical beads
Ionic Form as Shipped	Cl^-

	AMBERSEP™ 21K XLT	AMBERSEP™ 21K 16–20
Total Exchange Capacity	≥ 1.4 eq/L	≥ 1.2 eq/L
Water Retention Capacity	50 – 60%	50 – 58%
Particle Size		
Particle Diameter §	575 ± 50 µm	800 – 1300 µm
Uniformity Coefficient	≤ 1.1	
< 840 µm		≤ 10%
< 710 µm		≤ 2%
Whole Uncracked Beads	≥ 95%	≥ 90%
Swelling	$Cl^- \rightarrow OH^-$: 18 – 20%	$Cl^- \rightarrow OH^-$: 20%
Particle Density	1.08 g/ml.	1.08 g/ml.
Bulk Density, as Shipped	670 g/L	690 g/L

§ For additional particle size information, please refer to the Particle Size Distribution Cross Reference Chart (Form No. 177-01775).

Form No. 177-03691, Rev 0
January 2018

Suggested Operating Conditions**

Maximum Operating Temperature	
Cl^- Form	100°C (212°F)
OH^- Form	60°C (140°F)
pH Range	0 – 14
Bed Depth, min.	800 mm (2.6 ft)
Organic Loading	≤ 3 g $KMnO_4$/L resin

	AMBERSEP™ 21K XLT	AMBERSEP™ 21K 16–20
Flowrates		
Service	5 – 60 m/h (2 – 24 gpm/ft²)	5 – 50 m/h (2 – 20 gpm/ft²)
Backwash	See Figure 1	See Figure 1
Regeneration		
Chemical Injection		
Co-current	1 – 10 m/h (0.4 – 4 gpm/ft²)	1 – 10 m/h (0.4 – 4 gpm/ft²)
Counter-current	5 – 20 m/h (2 – 8 gpm/ft²)	
Displacement Rinse		
Co-current	1 – 10 m/h (0.4 – 4 gpm/ft²)	1 – 10 m/h (0.4 – 4 gpm/ft²)
Counter-current	5 – 20 m/h (2 – 8 gpm/ft²)	
Fast Rinse	5 – 60 m/h (2 – 24 gpm/ft²)	5 – 50 m/h (2 – 20 gpm/ft²)
Total Rinse Requirement	3 – 6 BV*	3 – 6 BV*
Regenerant	NaCl, Na_2CO_3, NaOH	
Temperature	Ambient or up to 50°C (122°F) for silica removal	

* 1 BV (Bed Volume) = 1 m³ solution per m³ resin or 7.5 gal per ft³ resin

Hydraulic Characteristics

Bed expansion of AMBERSEP™ 21K XLT and AMBERSEP 21K 16–20 Ion Exchange Resins as a function of backwash flowrate at 25°C (77°F) is shown in Figure 1.

Pressure drop data for AMBERSEP 21K XLT and AMBERSEP 21K 16–20 as a function of service flowrate at 25°C (77°F) is shown in Figure 2. Pressure drop data are valid at the start of the service run with clean water.

Figure 1: Backwash Expansion

Temperature = 25°C (77°F)

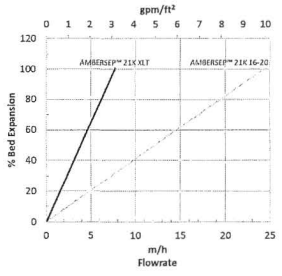
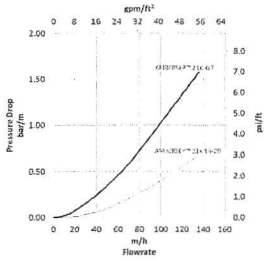


Figure 2: Pressure Drop

Temperature = 25°C (77°F)



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WARNING: Oxidizing agents such as nitric acid attack organic ion exchange resins under certain conditions. This could lead to anything from slight resin degradation to a violent exothermic reaction (explosion). Before using strong oxidizing agents, consult sources knowledgeable in handling such materials.

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Form No. 177-03691, Rev. 0
January 2018

CHECKLIST ITEMS[1]	YES	NO	N/A
GENERAL REQUIREMENTS			
1. If the calculation is being performed to a client procedure, is the procedure being used the latest revision? Client procedure is not used in this calculation. ENERCON QA procedures used throughout this project.	☐	☐	☒
2. Are the proper forms being used and are they the latest revision? Same format matching EPM017-CALC-001 was used for internal consistency	☐	☒	☐
3. Have the appropriate client review forms/checklists been completed? Client procedure is not used in this calculation. ENERCON QA procedures used throughout this project.	☐	☐	☒
4. Are all pages properly identified with a calculation number, calculation revision and page number consistent with the requirements of the client's procedure? Client procedure is not used in this calculation. ENERCON QA procedures used throughout this project.	☐	☐	☒
5. Is all information legible and reproducible?	☒	☐	☐
6. Is the calculation presented in a logical and orderly manner?	☒	☐	☐
7. Is there an existing calculation that should be revised or voided?	☐	☒	☐
8. Is it possible to alter an existing calculation instead of preparing a new calculation for this situation? No current ENERCON calculations exist that are similar to this calculation.	☐	☒	☐
9. If an existing calculation is being used for design inputs, are the key design inputs, assumptions and engineering judgments used in that calculation valid and do they apply to the calculation revision being performed.	☐	☒	☐
10. Is the format of the calculation consistent with applicable procedures and expectations?	☒	☐	☐
11. Were design input/output documents properly updated to reference this calculation? No ENERCON design inputs or outputs are affected by this calculation.	☐	☐	☒

CHECKLIST ITEMS[1]	YES	NO	N/A
12. Can the calculation logic, methodology and presentation be properly understood without referring back to the originator for clarification?	☒	☐	☐
OBJECTIVE AND SCOPE			
13. Does the calculation provide a clear concise statement of the problem and objective of the calculation?	☒	☐	☐
14. Does the calculation provide a clear statement of quality classification?	☒	☐	☐
15. Is the reason for performing and the end use of the calculation understood?	☒	☐	☐
16. Does the calculation provide the basis for information found in the plant's license basis? This calculation applies to a remediation site. No work performed in this calculation is applicable to a licensing basis.	☐	☐	☒
17. If so, is this documented in the calculation?	☐	☐	☒
18. Does the calculation provide the basis for information found in the plant's design basis documentation?	☐	☐	☒
19. If so, is this documented in the calculation?	☐	☐	☒
20. Does the calculation otherwise support information found in the plant's design basis documentation?	☐	☐	☒
21. If so, is this documented in the calculation?	☐	☐	☒
22. Has the appropriate design or license basis documentation been revised, or has the change notice or change request documents being prepared for submittal?	☐	☐	☒
DESIGN INPUTS			
23. Are design inputs clearly identified?	☒	☐	☐
24. Are design inputs retrievable or have they been added as attachments?	☒	☐	☐

	CHECKLIST ITEMS[1]	YES	NO	N/A
25.	If Attachments are used as design inputs or assumptions are the Attachments traceable and verifiable? MS Excel spreadsheet was used to perform the calculation. All equations are provided in the calculation.	☐	☐	☒
26.	Are design inputs clearly distinguished from assumptions?	☒	☐	☐
DESIGN INPUTS (Continued)				
27.	Does the calculation rely on Attachments for design inputs or assumptions? If yes, are the attachments properly referenced in the calculation?	☐	☒	☐
28.	Are input sources (including industry codes and standards) appropriately selected and are they consistent with the quality classification and objective of the calculation?	☒	☐	☐
29.	Are input sources (including industry codes and standards) consistent with the plant's design and license basis?	☐	☐	☒
30.	If applicable, do design inputs adequately address actual plant conditions?	☐	☐	☒
31.	Are input values reasonable and correctly applied?	☒	☐	☐
32.	Are design input sources approved? The Cimarron design is currently at 60% completion.	☐	☒	☐
33.	Does the calculation reference the latest revision of the design input source?	☒	☐	☐
34.	Were all applicable plant operating modes considered?	☐	☐	☒
ASSUMPTIONS				
35.	Are assumptions reasonable/appropriate to the objective?	☒	☐	☐
36.	Is adequate justification/basis for all assumptions provided?	☒	☐	☐
37.	Are any engineering judgments used?	☒	☐	☐

CHECKLIST ITEMS[1]	YES	NO	N/A
38. Are engineering judgments clearly identified as such? Engineering judgement applied to factors used in potential intake calculation using NUREG-1400 methodology.	☒	☐	☐
39. If engineering judgments are utilized as design inputs, are they reasonable and can they be quantified or substantiated by reference to site or industry standards, engineering principles, physical laws or other appropriate criteria?	☒	☐	☐
METHODOLOGY			
40. Is the methodology used in the calculation described or implied in the plant's licensing basis?	☐	☐	☒
41. If the methodology used differs from that described in the plant's licensing basis, has the appropriate license document change notice been initiated?	☐	☐	☒
42. Is the methodology used consistent with the stated objective?	☒	☐	☐
43. Is the methodology used appropriate when considering the quality classification of the calculation and intended use of the results?	☒	☐	☐
BODY OF CALCULATION			
44. Are equations used in the calculation consistent with recognized engineering practice and the plant's design and license basis?	☒	☐	☐
45. Is there reasonable justification provided for the use of equations not in common use? Equations applied in this evaluation are in common use in the industry.	☐	☐	☒
46. Are the mathematical operations performed properly and documented in a logical fashion?	☒	☐	☐
47. Is the math performed correctly?	☒	☐	☐
48. Have adjustment factors, uncertainties and empirical correlations used in the analysis been correctly applied?	☒	☐	☐
49. Has proper consideration been given to results that may be overly sensitive to very small changes in input? Results generated by calculations performed in this evaluation are not significantly affected by minor perturbations of variables.	☐	☐	☒

CHECKLIST ITEMS[1]	YES	NO	N/A
SOFTWARE/COMPUTER CODES			
50. Are computer codes or software languages used in the preparation of the calculation?	☐	☒	☐
51. Have the requirements of CSP 3.09 for use of computer codes or software languages, including verification of accuracy and applicability been met?	☐	☐	☒
SOFTWARE/COMPUTER CODES (Continued)			
52. Are the codes properly identified along with source vendor, organization, and revision level?	☐	☐	☒
53. Is the computer code applicable for the analysis being performed?	☐	☐	☒
54. If applicable, does the computer model adequately consider actual plant conditions?	☐	☐	☒
55. Are the inputs to the computer code clearly identified and consistent with the inputs and assumptions documented in the calculation?	☐	☐	☒
56. Is the computer output clearly identified?	☐	☐	☒
57. Does the computer output clearly identify the appropriate units? The output units are not identified in the output document. Tallies have been modified through multipliers and dose response functions. This process has been adequately documented within this calculation.	☐	☐	☒
58. Are the computer outputs reasonable when compared to the inputs and what was expected? Only basic functions and operations in Microsoft Excel 2013 were applied in this calculation.	☒	☐	☐
59. Was the computer output reviewed for ERROR or WARNING messages that could invalidate the results?	☐	☐	☒
RESULTS AND CONCLUSIONS			
60. Is adequate acceptance criteria specified? No acceptance criteria required for this evaluation.	☐	☐	☒

CHECKLIST ITEMS[1]	YES	NO	N/A
61. Are the stated acceptance criteria consistent with the purpose of the calculation, and intended use?	☐	☐	☒
62. Are the stated acceptance criteria consistent with the plant's design basis, applicable licensing commitments and industry codes, and standards?	☐	☐	☒
63. Do the calculation results and conclusions meet the stated acceptance criteria?	☐	☐	☒
64. Are the results represented in the proper units with an appropriate tolerance, if applicable?	☒	☐	☐
65. Are the calculation results and conclusions reasonable when considered against the stated inputs and objectives?	☒	☐	☐
66. Is sufficient conservatism applied to the outputs and conclusions?	☒	☐	☐
67. Do the calculation results and conclusions affect any other calculations? No ENERCON calculations are affected by this evaluation.	☐	☐	☒
68. If so, have the affected calculations been revised?	☐	☐	☒
69. Does the calculation contain any conceptual, unconfirmed or open assumptions requiring later confirmation?	☐	☒	☐
70. If so, are they properly identified?	☐	☐	☒
DESIGN REVIEW			
71. Have alternate calculation methods been used to verify calculation results?	☒	☐	☐

Note:

1. Where required, provide clarification/justification for answers to the questions in the space provided below each question. An explanation is required for any questions answered as "No' or "N/A".

Originator: J Maisler

Jay Maisler	4/4/2022
Print Name and Sign	Date

APPENDIX B

DOSE RATE NEAR URANIUM TREATMENT TRAIN (EPM017-CALC-001)

ENERCON Excellence—Every project. Every day.	CALCULATION COVER SHEET	CALC NO.	EPM017-CALC-001
		REV.	0
		PAGE NO.	1 of 15

Title:	Dose Rate Near Uranium Treatment Train	Client:	Cimarron Environmental Response Trust
		Project Identifier:	EPM017

Item	Cover Sheet Items	Yes	No
1	Does this calculation contain any open assumptions, including preliminary information, that require confirmation? (If **YES**, identify the assumptions.)	☐	☒
2	Does this calculation serve as an "Alternate Calculation"? (If **YES**, identify the design verified calculation.) **Design Verified Calculation No.** _____	☐	☒
3	Does this calculation supersede an existing Calculation? (If **YES**, identify the design verified calculation.) **Superseded Calculation No.** _____	☐	☒

Scope of Revision:
Initial Issue

Revision Impact on Results:
Initial Issue

Study Calculation ☐ **Final Calculation** ☒

Safety-Related ☐ **Non-Safety-Related** ☒

(Print Name and Sign)

Originator: Caleb Trainor	**Date: 12/21/2015**
Design Reviewer: John Hawkinson	**Date: 12/23/2015**
Approver: Jay Maisler, CHP	**Date: 12/23/2015**

9/28/2022 - Page numbering in Attachment C was corrected to show a total of 29 pages; no changes to actual calculation were made. JJM/RSO

ENERCON Excellence—Every project. Every day.	CALCULATION REVISION STATUS SHEET	CALC NO. EPM017-CALC-001
		REV. 0
		PAGE NO. 2 of 15

CALCULATION REVISION STATUS

REVISION	DATE	DESCRIPTION
0	12/21/2015	Initial Issue

PAGE REVISION STATUS

PAGE NO.	REVISION	PAGE NO.	REVISION

APPENDIX/ATTACHMENT REVISION STATUS

APPENDIX NO.	NO. OF PAGES	REVISION NO.	ATTACHMENT NO.	NO. OF PAGES	REVISION NO.

CALCULATION TABLE OF CONTENTS

ENERCON Excellence—Every project. Every day.	TABLE OF CONTENTS	CALC NO.	EPM017-CALC-001
		REV.	0

1.0 Purpose and Scope

The purpose of this calculation is to determine the radiation level surrounding a resin vessel array used to extract uranium contamination from groundwater at the Cimarron Site, hereby referred to as a uranium treatment train. The results of this calculation represent the highest anticipated dose rate an individual would be subject to while in the close vicinity of a uranium treatment train.

2.0 Summary of Results and Conclusions

The highest anticipated dose rate in the vicinity of a uranium treatment train is calculated to be 0.024 mrem/hr.

3.0 References

3.1 MCNP6-1.0, Revision 00, MCNP6 Version 1.0 Acceptance Report, November 2014
3.2 LA-CP-13-000634, MCNP6 User's Manual, Version 1.0, May 2013
3.3 MicroShield 6.20
3.4 DTS-WPS-M-3100, AVANTech Drawing, '48" Process Vessel', Rev B, January 10, 2000
3.5 Vessel Use and U235 Accum Calc_DRAFT_16Oct15.xlsx, Vessel loading analysis provided as an Excel spreadsheet
3.6 PNNL-15870, Rev. 1, Compendium of Material Composition Data for Radiation Transport Modeling, March 4, 2011
3.7 Email correspondence with Kurion representative (attached)
3.8 BA1 Isotopic Data for Enercon.xlsx, Isotopic abundance of Uranium source provided as an Excel spreadsheet
3.9 NUREG/BR-0150, Vol. 1, Rev. 4, RTM-96 Response Technical Manual, March 1996
3.10 Introduction to Health Physics, 4th ed., Herman Cember and Thomas E. Johnson, McGraw-Hill, 2009.

4.0 Assumptions

4.1 The resin vessel is approximated as a perfect right circular cylinder, maintaining the actual volume and clad thickness of the vessel. This is a minor simplification made for modeling purposes and is insignificant to the final dose rate.

4.2 The resin vessel is modeled as stainless steel 304. At the time of this calculation, the vessel was known to be composed of stainless steel, but the specific alloy was

unknown. Stainless steel 304 was chosen as it is a popular alloy for nuclear vessels. The specific alloy of stainless steel has insignificant impact on shielding.

4.3 The mass of uranium assumed to be present in the resin vessels is based off of a draft calculation provided for this study. The calculation is performed in Excel spreadsheet "Vessel Use and U235 Accum Calc_DRAFT_16Oct15.xlsx" (Reference 3.5). From this calculation, the largest expected mass accumulation of uranium is 24,055 g of uranium at Burial Area 1 (BA1) after the first full loading. The theoretical capacity of the resin used for the calculation is 29,589.01 grams.

The vessel loading calculations use an enrichment of 2% by weight. This value was chosen as a conservative bound for the actual 1.66% enrichment in order to remain conservative with respect to materials controls. The resin loading is not dependent on enrichment and the input values used in this calculation are not affected by this assumption.

4.4 The lead vessel is modeled assuming it is completely full to its theoretical capacity, ready to be removed and replaced with the lag vessel. The lag vessel is assumed to accumulate one week's worth of uranium, due to a one week period between sampling intervals. The accumulation in the first week of loading in the lead vessel is used to define one week's worth of accumulation. This is conservative as the uranium loading rate decreases with time; as the concentration of contaminants in the fluid passed through the vessels decreases, the resin removes a lower volume of uranium. The polishing vessel is assumed to provide negligible dose rate contribution due to the significant distance and low accumulation of uranium.

4.5 The uranium is assumed to be evenly distributed through the resin vessel.

4.6 The DOWEX-1 resin is approximated as polystyrene. For shielding purposes, this media is roughly equivalent. The mass of resin used is provided in Reference 3.5. The resin is assumed to fill the vessel and is homogenized, while maintaining mass, for modeling purposes.

4.7 The shielding provided by the vessel clad and resin are modeled. The shielding which would be provided by the water and additional piping inside the vessel has been omitted. Miscellaneous piping and structures outside the vessels are also omitted. The lack of additional shielding provides a conservative estimate, which results in a higher estimation of dose rate.

4.8 All alpha and beta particles are shielded by the 3/8 inch stainless steel walls of the resin vessel and do not contribute to the final dose rate.

4.9 The spontaneous fission rate for the uranium isotopes is sufficiently low (3.5 n/s per kg U-234, 0.31 n/s per kg U-235 and 7.0 n/s per kg U-238) that the neutron dose rate can be disregarded.

5.0 Design Inputs

5.1 Resin Vessel Properties

The following properties are taken from Reference 3.4.

Vessel Diameter=48 in
Volume=54.5 ft^3

The following properties are taken from Reference 3.7.

Vessel Clad=3/8 in
Vessel Spacing=48 in (96 in on center)

5.2 Source Characteristics

The following source characteristics are taken from Reference 3.5

Media Capacity=39.52 g U/kg
Media Mass used=748.7 kg

The following isotopic abundances by mass percent are taken from Reference 3.8.

U_{234}= 0.01
U_{235}= 1.66
U_{238}= 98.33

The following specific activities are taken from Table E-4 of Reference 3.9.

U_{234}= 6.19 E+03 µCi/g
U_{235}= 2.14 E+00 µCi/g
U_{238}= 3.33 E-01 µCi/g

5.3 Material Definitions

The following material definitions are taken from Reference 3.6.

Table 5.1: Composition of Polystyrene (Density homogenized to 0.485139 g/cc)

Nuclide	Weight Fraction
Hydrogen	0.077421
Carbon	0.922579

Table 5.2: Composition of Stainless Steel 304 (Density of 8.0 g/cc)

Nuclide	Weight Fraction
Carbon	0.000400
Silicon	0.005000
Phosphorus	0.000230
Sulphur	0.000150
Chromium	0.190000
Manganese	0.010000
Iron	0.701730
Nickel	0.092500

5.4 Resin Vessel Loading

The first week's loading of the resin vessel is 3,706 grams of uranium. After the first full loading, the vessel contains 24,055 grams of uranium. These values are taken from Reference 3.5.

6.0 Methodology

6.1 Source Term

The activity by isotope is determined using the mass abundance of each isotope from Reference 3.8 and applying the specific activities from Reference 3.9. These activities are entered into MicroShield 6.20 (Reference 3.3) to produce a list of gamma emission energies and activity in photons per second. These energies correspond to the energy bins defined in the MCNP6 (Reference 3.2) source term distribution. The activity in photons per second correspond to the energy distribution probabilities defined in the MCNP6 source term.

6.2 Shielding Evaluation

The MCNP6 code is used to model the resin vessel and cylindrical source term inside the vessel. The interior of the vessel is modeled as homogenized polystyrene. Outside of the vessel is comprised of dry air, characteristic of near sea level. Two point detectors used, one located mid height of the vessel, 12 inches (30.48 cm) from the outer surface of the vessel and another located 96 inches

(283.84 cm) offset laterally. The first detector represents the dose rate contribution from the modeled vessel (the lead vessel), the second detector represents the dose rate contribution from the lag vessel. Figures 6.1 and 6.2 are representative of the MCNP6 input.

Figure 6.1: Z-plane of Modeled Geometry

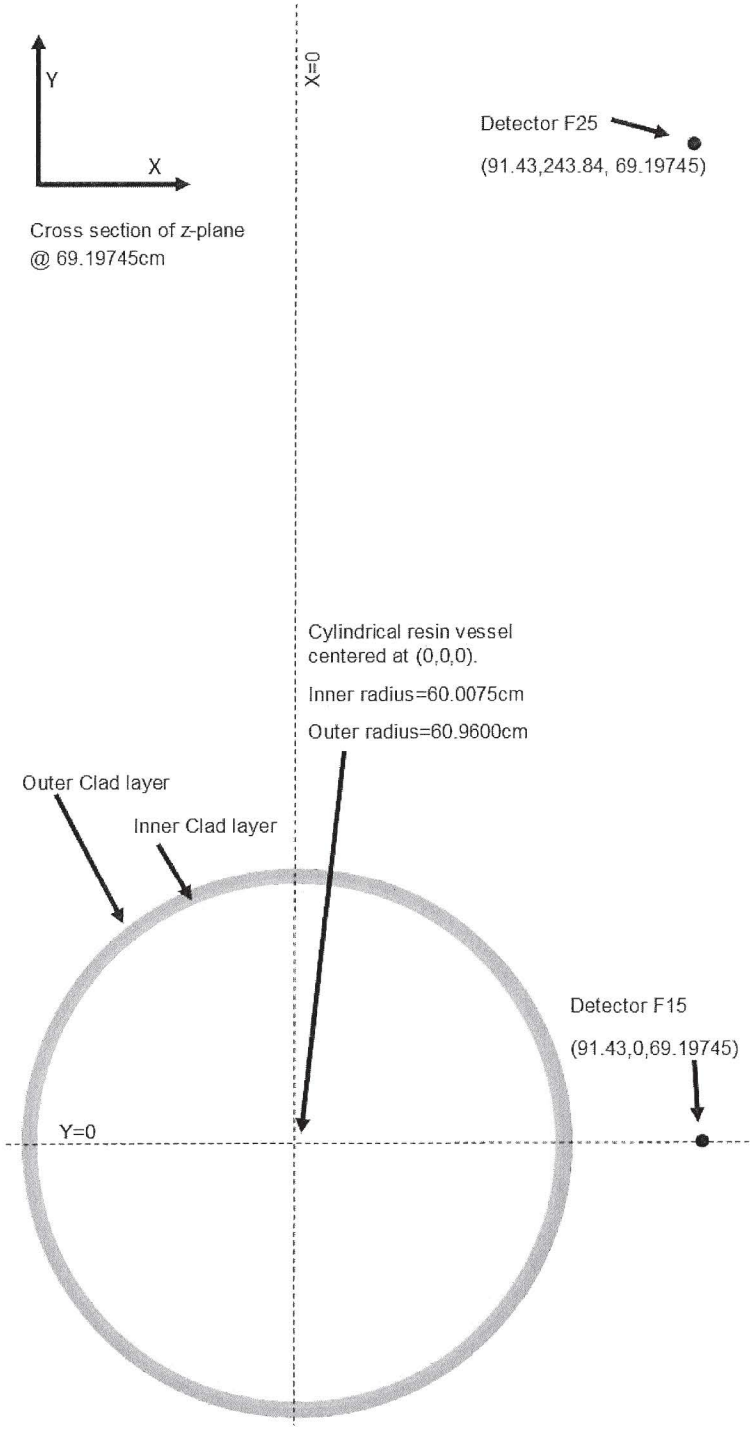
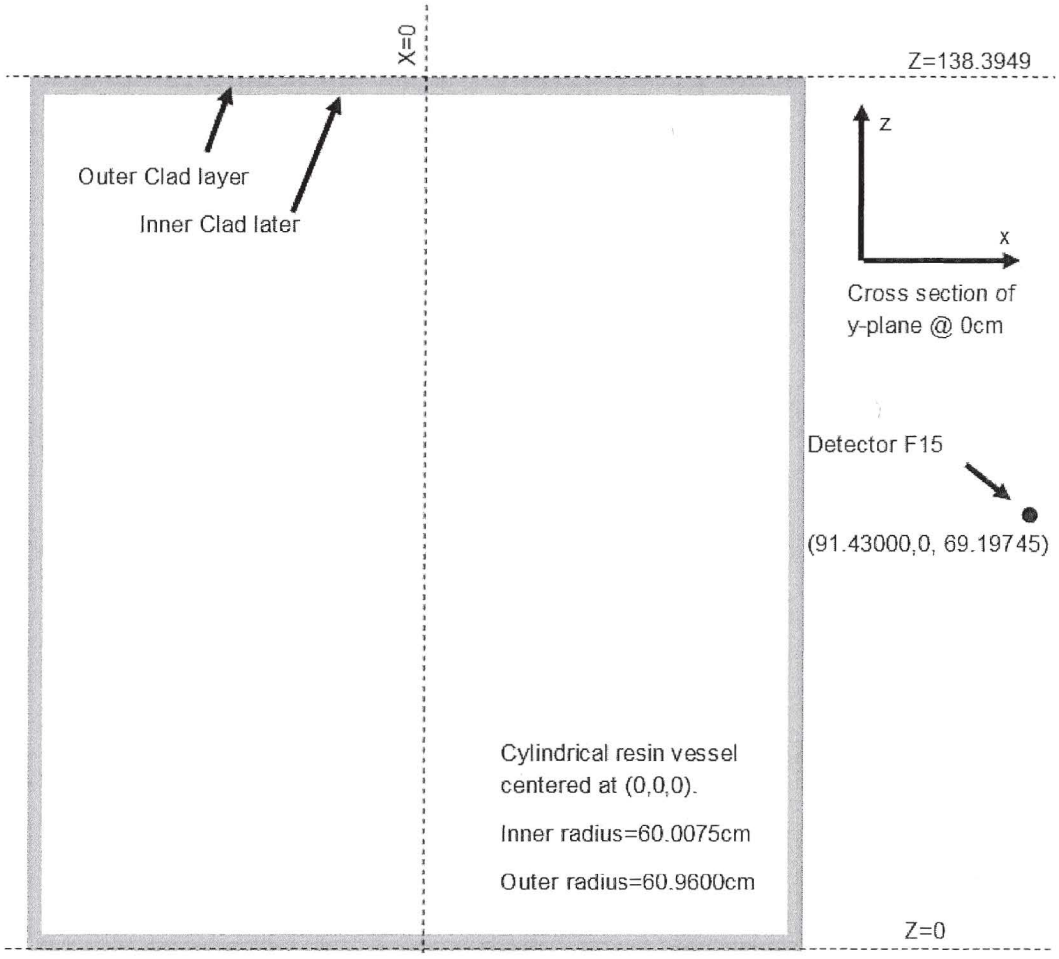


Figure 6.2: Y-Plane of Modeled Geometry



7.0 Calculation

7.1 Source Development

The activity of each isotope is found using the Equation 7.1.

$$A_i = U_i(g) * SA_i\left(\frac{\mu Ci}{g}\right)$$

Equation 7.1

Where A_i is equal to activity of each isotope, in µCi, U_i is the mass of uranium isotope i, and SA_i is equal to the specific activity of uranium isotope i.

The mass of each isotope in a full resin vessel is found using Equation 7.2.

$$U_i(g) = Media\ capacity\left(\frac{U(g)}{kg}\right) * Media\ mass\ (kg) * w_i$$

Equation 7.2

Where w_i represents the isotopes abundance by weight percent. Solving for the three isotopes;

$$U_{234}(g) = 39.52\left(\frac{U(g)}{kg}\right) * 748.7\ (kg) * 0.0001 = 2.958\ g\ U_{234}$$

$$U_{235}(g) = 39.52\left(\frac{U(g)}{kg}\right) * 748.7\ (kg) * 0.0166 = 491.177\ g\ U_{235}$$

$$U_{238}(g) = 39.52\left(\frac{U(g)}{kg}\right) * 748.7\ (kg) * 0.9833 = 29094.873\ g\ U_{238}$$

Using these values, and the specific activities defined in Design Input 5.2, Equation 7.1 is solved for the three isotopes;

$$A_{U_{234}} = 2.958\ (g) * 6190\ \left(\frac{\mu Ci}{g}\right) = 1.83 * 10^4 \mu Ci$$

$$A_{U_{235}} = 491.177\ (g) * 2.140\ \left(\frac{\mu Ci}{g}\right) = 1.05 * 10^3 \mu Ci$$

$$A_{U_{238}} = 29094.873(g) * 0.333\ \left(\frac{\mu Ci}{g}\right) = 9.69 * 10^3 \mu Ci$$

With the activities determined, MicroShield 6.20 is used to define gamma emission energies and emission frequencies. The MicroShield output is listed in Table 7.1.

Table 7.1: Gamma Emissions from Uranium source

Energy (MeV)	Activity (Photons/sec)
0.013	7.112E+07
0.013	3.166E+07
0.013	1.201E+07
0.0532	7.990E+05
0.0664	3.478E+05
0.0727	4.273E+04
0.09	1.060E+06
0.0933	1.732E+06
0.105	8.014E+05
0.1091	5.827E+05
0.12	5.827E+04
0.1214	2.712E+05
0.1408	8.547E+04
0.1438	4.079E+06
0.1633	1.826E+06
0.1827	1.554E+05
0.1857	2.098E+07
0.1904	3.572E+05
0.1949	2.292E+05
0.2021	3.885E+05
0.2053	1.826E+06
0.2214	3.885E+04

7.2 MCNP6 Input Development

7.2.1 Geometry Specifications

The only resin vessel modeled is the lead vessel. Since the lead vessel accumulates the most uranium, the area of interest is directly in front of the lead vessel. A detector is placed halfway up the vessel, 12 inches (30.48 cm) from the outer surface of the vessel. To model the dose rate contribution from the lag vessel, a second detector is placed at the same height, offset laterally by 96 inches (283.84 cm). The inner and outer clad layers represent the vessel clad, seperated into two layers for variance reduction purposes.

The geometry between the modeled vessel and the second detector is equivalent to the geometry between the lag vessel and the dose rate point of interest. The second detector tally will be scaled accordingly to represent the source term which would be present in the lag vessel.

The lead vessel is modeled using two concentric right circular cylinders with a separation of 3/8 inches around the top, sides, and bottom. The dimensions of the interior cylinder are calculated, preserving the defined vessel volume and diameter.

$$Vol = h_i * \pi {r_i}^2$$

Equation 7.4

Where;

$$Vol = 54.5\ ft^3$$

$$r_i = \frac{D - (2 * clad)}{2} = \frac{48\ in. - (2 * \frac{3}{8} in.)}{2} = 23.625\ in$$

Solving for h_i;

$$h_i = \frac{54.5\ ft^3 * 1728\ \frac{in^3}{ft^3}}{3.14159 * (23.625\ in)^2} = 53.7362\ in$$

The clad thickness is accounted for in the dimensions for the outer cylinder in the following equations.

$$h_o = h_i + 2 * Clad$$

$$= 53.7362\ in + 2 * {}^3/_8\ in = 54.4862\ in$$

$$r_o = r_i + Clad$$

$$= 23.625\ in + {}^3/_8\ in = 24\ in$$

For variance reduction purposes, a third right circular cylinder is included. This cylinder divides the vessel clad into two equal layers, shown as the "inner clad layer" and "outer clad layer" in Figures 6.1 and 6.2. The following excerpt from the MCNP6 input deck shows the cell and surface cards which define the geometry described above.

Figure 7.1: Cell and Surface Cards from Input Deck

```
c Cells
10 2 -0.485139 -100          imp:p=1          $Resin (polystyrene) inside vessel
50 1 -8 -10 100              imp:p=2          $SS 304 layer
11 1 -8 -101 10 100          imp:p=4          $SS 304 vessel clad
30 0 -300 101                imp:p=8          $Void
40 0 300                     imp:p=0          $Boundary

c Surfaces
100 RCC 0 0 0.9525 0 0 136.4899 60.0075       $Interior cylinder of resin vessel,
10  RCC 0 0 0.47625 0 0 136.96615 60.48375     $Layer used for variance reduction
101 RCC 0 0 0 0 0 138.3949 60.96               $Outer cylinder of resin vessel
102 S 91.43 0 69.19745 20                     $Detector 1 Void
103 S 91.43 243.84 69.19745 20                $Detector 2 Void
300 RPP -65 100 -65 250 0 150                 $Bounding Area
```

7.2.2 Source Definition

The source is defined as a photon source, evenly distributed within a right circular cylinder equivalent to the interior cylinder of the resin vessel. The values in Table 7.1 are used to define the energy and probability of gamma particles in this area. The terms axs, pos, rad, and ext define the vector, base position, radius and height of the source term respectively.

Figure 7.2: Source Definition from Input Deck

```
c Source Definition
SDEF PAR=P axs=0 0 1 pos=0 0 0 rad=d1 ext=d2 erg=d3 cel=10        $Cylindrical source located inside inner cylinder
SI1 0 60.0075
SI2 0.9525 137.4424
SI3 L 0.13 .013 .013 .0532 .0664 .0727 .09 .0933 .105             $Discrete energy distribution based on gamma energies from MicroShield
     .1091 .12 .1214 .1408 .1438 .1633 .1827 .1857 .1904
     .1949 .2021 .2053 .2214
SP3  7.112e7 3.166e7 1.201e7 7.990e5 3.478e5 4.273e4 1.060e6      $Gamma emission frequency from MicroShield
     1.732e6 8.014e5 5.827e5 5.827e4 2.712e5 8.547e4 4.079e6
     1.826e6 1.554e5 2.098e7 3.572e5 2.292e5 3.885e5 1.826e6
     3.885e4
```

7.2.3 Tally Definitions

Two tallies are used to find the total dose rate. The first tally is located mid height and one foot away from the modeled lead resin vessel. The second tally is located in the equivalent position on the lag resin vessel (not modeled). Using these tallies, the total dose rate is equal to the summation of the first and second tally.

A tally multiplier of 1.504 E+08 is applied to the first tally. This value is equal to the summation of column 2 of Table 7.1 and scales the results to the source strength. A tally multiplier of 2.568 E+07 is applied to the second tally. This is approximately 15% of the source strength used for the first tally. This tally represents the dose rate contribution of the lag vessel, acting as the primary resin vessel for one week's time per Assumption 4.4. A ratio of 15%

is equivalent to the ratio of the first weeks vessel loading compared to a fully loaded vessel. The values from Design Input 5.4 are used to determine this ratio.

An MCNP6 dose response function is also applied to all tallies. This function sorts the tallies into energy bins, applies dose conversion factors from ANSI 6.1.1-1977, and sums all the bins. Due to the tally multiplier and dose function, the output of the tallies is in rem per hour. Figure 7.3 shows the tally definitions from the input deck.

Figure 7.3: Tally Definitions from Input Deck

```
c Tally Definition
F15:p 91.43 0 69.19745 10                    $Point Detector, mid height of vessel, one foot away
F25:p 91.43 243.84 69.19745 10               $Point Detector, mid height of vessel, one foot away, 96" offset
c                                            $equivalent to dose contribution from lag vessel
FM15 1.5045072E8                             $Tally multiplier equivalent to summation of all gamma emission frequencies
FM25 2.2567608e7                             $Source scaled to one week of buildup in lag vessel
c
c    ANSI 6.1.1-1977 Gamma Flux to Dose Conversion Factors, using US units
c    (rem/hr)/(photons/cm2-s)
df0  IU 1 IC 20                              $Seperates tally into energy bins, applies conversion factors and sums
```

7.3 Dose Rates

The estimated dose rates present near the uranium treatment train are presented in Table 7.2.

Table 7.2: Dose Rate

Tank Contribution	Equivalent Tally	Dose Rate (rem/hr)
Lead tank	F15	2.392 E-05
Lag tank	F25	4.974 E-07
	Total Dose Rate	2.442 E-05 (.02442 mrem/hr)

8.0 Computer Software

MCNP6 is used for shielding analysis in this calculation. It is verified and validated for use (Reference 3.1).

MicroShield 6.20 is used for calculating gamma emission energies and frequencies.

Attachment A

MCNP6 Input

```
Cimmarron Resin Bed Dose Estimation
c Cells
10 2 -0.485139 -100          imp:p=1                          $Resin (polystyrene) inside vessel
50 1 -8 -10 100              imp:p=2                          $SS 304 layer
11 1 -8 -101 10 100          imp:p=4                          $SS 304 vessel clad
30 0 -300 101                imp:p=8                          $Void
40 0 300                     imp:p=0                          $Boundary

c Surfaces
100 RCC 0 0 0.9525 0 0 136.4899 60.0075                       $Interior cylinder of resin vessel, 3/8" (0.9525cm) off of floor to accomodate vessel clad
10  RCC 0 0 0.47625 0 0 136.96615 60.48375                    $Layer used for variance reduction
101 RCC 0 0 0 0 0 138.3949 60.96                              $Outer cylinder of resin vessel
102 S 91.43 0 69.19745 20                                     $Detector 1 Void
103 S 91.43 243.84 69.19745 20                                $Detector 2 Void
300 RPP -65 100 -65 250 0 150                                 $Bounding Area

c Data
Mode P
nps 10e6
c
c Material Definitions
M1   6000 -0.000400 14000 -0.005000 15000 -0.000230           $SS 304
     16000 -0.000150 24000 -0.190000 25000 -0.010000
     26000 -0.701730 28000 -0.092500
M2  1000 -0.077421 6000 -0.922579                             $Polystyrene
c
c Source Definition
SDEF PAR=P axs=0 0 1 pos=0 0 0 rad=d1 ext=d2 erg=d3 cel=10    $Cylindrical source located inside inner cylinder
SI1 0 60.0075
SI2 0.9525 137.4424
SI3 L 0.13 .013 .013 .0532 .0664 .0727 .09 .0933 .105         $Discrete energy distribution based on gamma energies from MicroShield
     .1091 .12 .1214 .1408 .1438 .1633 .1827 .1857 .1904
     .1949 .2021 .2053 .2214
SP3  7.112e7 3.166e7 1.201e7 7.990e5 3.478e5 4.273e4 1.060e6  | $Gamma emission frequency from MicroShield
     1.732e6 8.014e5 5.827e5 5.827e4 2.712e5 8.547e4 4.079e6
     1.826e6 1.554e5 2.098e7 3.572e5 2.292e5 3.885e5 1.826e6
     3.885e4
c
c Tally Definition
F15:p 91.43 0 69.19745 10                                     $Point Detector, mid height of vessel, one foot away
F25:p 91.43 243.84 69.19745 10                                $Point Detector, mid height of vessel, one foot away, 96" offset
c                                                             $equivalent to dose contribution from lag vessel
FM15 1.5045072E8                                              $Tally multiplier equivalent to summation of all gamma emission frequencies
FM25 2.2567608e7                                              $Source scaled to one week of buildup in lag vessel
c
c   ANSI 6.1.1-1977 Gamma Flux to Dose Conversion Factors, using US units
c   (rem/hr)/(photons/cm2-s)
df0  IU 1 IC 20                                               $Seperates tally into energy bins, applies conversion factors and sums
```

Attachment B
Email Correspondence

Caleb Trainor

From: Ja-Kael Luey <jluey@kurion.com>
Sent: Thursday, December 03, 2015 10:38 AM
To: Caleb Trainor
Cc: Charles Beatty; jlux@envpm.com; Ja-Kael Luey
Subject: RE: <EXTERNAL> RE: Resin Vessel Dimensions

Caleb,

If you are doing dose calculations use the following in lieu of specific information (which may be proprietary from AVANTech but I have not had a chance to check).

- Wall Thickness – 3/8". This is the wall thickness for Kurion's standard 36-inch vessel that has a similar working pressure. It is possible the AVANTech wall thickness is thicker so your dose calculation will be bounding. For dose purposes and alpha, not sure if the liner material will make much of a difference in your calculation.
- Spacing: DWG-M110 is the drawing that I thought had the separation distance on it. Looks like this was not a critical element to have on the drawing; however, based on the scale the distance is edge-to-edge as the gap looks to be the same as the ISM Vessel (which is 48-inches).

Ja-Kael

From: Caleb Trainor [mailto:ctrainor@enercon.com]
Sent: Thursday, December 03, 2015 7:00 AM
To: Ja-Kael Luey <jluey@kurion.com>
Subject: RE: <EXTERNAL> RE: Resin Vessel Dimensions

Thank you for the information, this is much closer to what I need. I am still left with a few questions however. Outer diameter is listed but no inner diameter; a liner is also mentioned but again no thickness or material. I assume the 48" spacing is edge to edge, or 96" on center, correct? It will be easy for me to calculate dose for any other spacing you give me, but a change in spacing is ultimately a crit safety concern, not dose.

From: Ja-Kael Luey [mailto:jluey@kurion.com]
Sent: Wednesday, December 02, 2015 11:22 PM
To: Caleb Trainor <ctrainor@enercon.com>
Cc: Charles Beatty <cbeatty@enercon.com>; jlux@envpm.com; Ja-Kael Luey <jluey@kurion.com>
Subject: <EXTERNAL> RE: Resin Vessel Dimensions

Caleb,

Please find attached the drawing that is the basis for the ISM Vessels used in the design. The spacing between the vessels is 48" from the General Assembly drawing (I can provide the specific reference when I am back in the office Thursday if you do not have the full set). This separation is based on meetings held with Enercon and not a specific-cited document. From a spacing standpoint it would be better if this could be smaller, especially for the BA Unit since it is fit into an enclosure.

Ja-Kael

From: Caleb Trainor [mailto:ctrainor@enercon.com]
Sent: Wednesday, December 02, 2015 6:34 AM

To: Ja-Kael Luey <jluey@kurion.com>
Cc: Charles Beatty <cbeatty@enercon.com>; jlux@envpm.com
Subject: Resin Vessel Dimensions

In support of dose calculations off the resin vessels, I am in need of the vessel dimensions, vessel material, and spacing between the vessels. The design drawings I have so far only detail the piping, and not the vessels. Can you help me get a hold of this information?

Caleb Trainor
Emergency Preparedness Engineer
(o) (813) 962-1800 ext. 207

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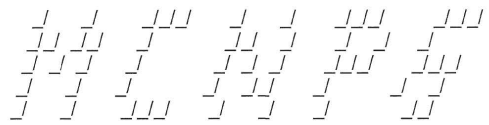
Attachment C

MCNP6 Output

Code Name & Version = MCNP6, 1.0



```
1mcnp    version 6    ld=05/08/13              12/15/15 16:46:08
 ******************************************************************************                probid =  12/15/15 16:46:08
 i=cim.txt o=cim.out tasks 8


 warning.  Physics models disabled.
      1-     Cimmarron Resin Bed Dose Estimation
      2-     c Cells
      3-     10 2 -0.485139 -100       imp:p=1                    $Resin (poly
      4-     50 1 -8 -10 100           imp:p=2                  $SS 304 laye
      5-     11 1 -8 -101 10 100       imp:p=4                    $SS 304 vess
      6-     30 0 -300 101             imp:p=8                  $Void
      7-     40 0 300                imp:p=0                  $Boundary
      8-
      9-     c Surfaces
     10-     100 RCC 0 0 0.9525 0 0 136.4899 60.0075                $Interior cy
     11-     10  RCC 0 0 0.47625 0 0 136.96615 60.48375               $Layer used
     12-     101 RCC 0 0 0 0 0 138.3949 60.96                     $Outer cylin
     13-     102 S 91.43 0 69.19745 20                          $Detector 1
     14-     103 S 91.43 243.84 69.19745 20                       $Detector 2
     15-     300 RPP -65 100 -65 250 0 150                        $Bounding Ar
     16-
     17-     c Data
     18-     Mode P
     19-     nps 10e6
     20-     c
     21-     c Material Definitions
     22-     M1  6000 -0.000400 14000 -0.005000 15000 -0.000230          $SS 304
     23-         16000 -0.000150 24000 -0.190000 25000 -0.010000
     24-         26000 -0.701730 28000 -0.092500
     25-     M2  1000 -0.077421 6000 -0.922579                    $Polystyrene
     26-     c
     27-     c Source Definition
```

```
28-     SDEF PAR=P axs=0 0 1 pos=0 0 0 rad=d1 ext=d2 erg=d3 cel=10        $Cylindrical
29-     SI1 0 60.0075
30-     SI2 0.9525 137.4424
31-     SI3 L 0.13 .013 .013 .0532 .0664 .0727 .09 .0933 .105            $Discrete en
32-        .1091 .12 .1214 .1408 .1438 .1633 .1827 .1857 .1904
33-        .1949 .2021 .2053 .2214
34-     SP3  7.112e7 3.166e7 1.201e7 7.990e5 3.478e5 4.273e4 1.060e6     $Gamma emiss
35-        1.732e6 8.014e5 5.827e5 5.827e4 2.712e5 8.547e4 4.079e6
36-        1.826e6 1.554e5 2.098e7 3.572e5 2.292e5 3.885e5 1.826e6
37-        3.885e4
38-     c
39-     c Tally Definition
40-     F15:p 91.43 0 69.19745 10                        $Point Detec
41-     F25:p 91.43 243.84 69.19745 10                     $Point Detec
42-     c                                          $equivalent
43-     FM15 1.5045072E8                                 $Tally multi
44-     FM25 2.2567608e7                                 $Source scal
45-     c
46-     c  ANSI 6.1.1-1977 Gamma Flux to Dose Conversion Factors, using US units
47-     c  (rem/hr)/(photons/cm2-s)
48-     df0  IU 1 IC 20                                  $Seperates t

surface     100.2 and surface      10.2 are the same.      10.2 will be deleted.

surface     101.3 and surface      300.6 are the same.     300.6 will be deleted.

 comment.          2 surfaces were deleted for being the same as others.

 warning.    1 materials had unnormalized fractions. print table 40.
1cells                                                          print table 60

                        atom       gram                              photon
          cell     mat  density    density     volume       mass          pieces importance
```

```
    1      10      2 4.48803E-02 4.85139E-01 1.54405E+06 7.49079E+05       1 1.0000E+00
    2      50      1 8.76841E-02 8.00000E+00 3.00795E+04 2.40636E+05       1 2.0000E+00
    3      11      1 8.76841E-02 8.00000E+00 4.15670E+04 3.32536E+05       1 4.0000E+00
    4      30      0 0.00000E+00 0.00000E+00 0.00000E+00 0.00000E+00       0 8.0000E+00
    5      40      0 0.00000E+00 0.00000E+00 0.00000E+00 0.00000E+00       0 0.0000E+00

total                                1.61570E+06 1.32225E+06

 warning. surface     102.0 is not used for anything.

 warning. surface     103.0 is not used for anything.

  minimum source weight = 1.0000E+00    maximum source weight = 1.0000E+00

*****************************************************
* Random Number Generator  =                1 *
* Random Number Seed       =      19073486328125 *
* Random Number Multiplier =      19073486328125 *
* Random Number Adder      =                0 *
* Random Number Bits Used  =               48 *
* Random Number Stride     =           152917 *
*****************************************************


 comment.  threading will be used when possible in portions of mcnp6.

 comment.  threading will be used for n/p/e table physics.

 comment.  threading will generally not be used for model physics.

      4 warning messages so far.
1cross-section tables                                          print table 100
   XSDIR used: C:\MCNP\MCNP_DATA/xsdir_mcnp6.1
```

table length

tables from file xdata/mcplib84

1000.84p	1974	Update of MCPLIB04 Photon Compton Broadening Data For MCNP5 see LA-UR-	12-00018	01/03/12
6000.84p	3228	Update of MCPLIB04 Photon Compton Broadening Data For MCNP5 see LA-UR-	12-00018	01/03/12
14000.84p	4868	Update of MCPLIB04 Photon Compton Broadening Data For MCNP5 see LA-UR-	12-00018	01/03/12
15000.84p	4574	Update of MCPLIB04 Photon Compton Broadening Data For MCNP5 see LA-UR-	12-00018	01/03/12
16000.84p	4730	Update of MCPLIB04 Photon Compton Broadening Data For MCNP5 see LA-UR-	12-00018	01/03/12
24000.84p	5758	Update of MCPLIB04 Photon Compton Broadening Data For MCNP5 see LA-UR-	12-00018	01/03/12
25000.84p	5674	Update of MCPLIB04 Photon Compton Broadening Data For MCNP5 see LA-UR-	12-00018	01/03/12
26000.84p	5794	Update of MCPLIB04 Photon Compton Broadening Data For MCNP5 see LA-UR-	12-00018	01/03/12
28000.84p	5902	Update of MCPLIB04 Photon Compton Broadening Data For MCNP5 see LA-UR-	12-00018	01/03/12

total 42502

maximum photon energy set to 100.0 mev (maximum electron energy)

tables from file xdata/el03

1000.03e	2329	6/6/98
6000.03e	2333	6/6/98
14000.03e	2339	6/6/98
15000.03e	2339	6/6/98
16000.03e	2339	6/6/98
24000.03e	2345	6/6/98
25000.03e	2345	6/6/98
26000.03e	2345	6/6/98
28000.03e	2347	6/6/98

1particles and energy limits print table 101

particle cutoff	maximum particle	smallest table	largest table	always use table	always use model

```
 particle type      energy      energy      maximum     maximum     below       above

  2 p  photon    1.0000E-03  1.0000E+02  1.0000E+05  1.0000E+05  1.0000E+36  1.0000E+36
  3 e  electron  1.0000E-03  1.0000E+02  1.0000E+02  1.0000E+02  1.0000E+36  1.0000E+36


 warning. material       1 has been set to a conductor.

************************************************************************************************************************

dump no.   1 on file runtpe    nps =         0    coll =          0    ctm =       0.00   nrn =
  0

      5 warning messages so far.

det    t      wgt      psc      amfp     ddetx    radius     erg      cell      nps  nch  p          nrn     ipsc
 1 4.2865E-04 2.5000E-01 1.4603E+01 3.0014E-01 3.1687E+01 1.0000E+01 1.0341E-01  11     29113   6            117      6

det    t      wgt      psc      amfp     ddetx    radius     erg      cell      nps  nch  p          nrn     ipsc
 1 4.6727E-04 2.5000E-01 2.9341E+01 6.9818E-01 3.5255E+01 1.0000E+01 1.0449E-01  11     37886   2             36      6

det    t      wgt      psc      amfp     ddetx    radius     erg      cell      nps  nch  p          nrn     ipsc
 1 4.3285E-04 2.5000E-01 2.9375E+01 7.7426E-01 3.5284E+01 1.0000E+01 1.0449E-01  11     37886   4             67      6

det    t      wgt      psc      amfp     ddetx    radius     erg      cell      nps  nch  p          nrn     ipsc
 1 7.1396E-04 2.5000E-01 2.9074E+01 3.7427E-01 3.3383E+01 1.0000E+01 1.8570E-01  11     99391   3             45      6

det    t      wgt      psc      amfp     ddetx    radius     erg      cell      nps  nch  p          nrn     ipsc
 1 5.0577E-04 2.5000E-01 7.5862E+01 1.0924E+00 4.4741E+01 1.0000E+01 1.3005E-01  11     202920  4             79      6

det    t      wgt      psc      amfp     ddetx    radius     erg      cell      nps  nch  p          nrn     ipsc
 1 5.2451E-04 2.5000E-01 1.8136E+01 4.9620E-02 3.6183E+01 1.0000E+01 1.3000E-01  11     266011  8            111      6

det    t      wgt      psc      amfp     ddetx    radius     erg      cell      nps  nch  p          nrn     ipsc
```

2 7.3518E-05 2.5000E-01 8.4093E+01 8.6295E-02 2.0433E+02 1.0000E+01 1.1394E-01 11 267347 4 66 6

det t wgt psc amfp ddetx radius erg cell nps nch p nrn ipsc
1 1.6404E-03 1.0000E+00 2.5645E+02 2.3748E+00 4.8113E+01 1.0000E+01 1.7341E-01 10 275363 3 54 6

det t wgt psc amfp ddetx radius erg cell nps nch p nrn ipsc
1 1.4661E-03 2.5000E-01 6.2765E+01 5.1569E-01 3.1892E+01 1.0000E+01 1.1387E-01 11 289480 3 65 6

det t wgt psc amfp ddetx radius erg cell nps nch p nrn ipsc
1 4.7999E-04 2.5000E-01 4.1311E+01 1.1433E+00 3.3039E+01 1.0000E+01 9.5904E-02 11 295930 3 60 6

det t wgt psc amfp ddetx radius erg cell nps nch p nrn ipsc
1 5.3206E-03 2.5000E-01 1.4027E+02 7.2495E-02 3.1234E+01 1.0000E+01 1.8570E-01 11 313718 3 41 6

det t wgt psc amfp ddetx radius erg cell nps nch p nrn ipsc
2 7.5842E-05 2.5000E-01 8.3107E+01 1.0047E-02 2.0776E+02 1.0000E+01 1.0709E-01 11 385577 4 71 6

det t wgt psc amfp ddetx radius erg cell nps nch p nrn ipsc
2 1.4009E-04 2.5000E-01 2.5325E+02 4.0166E-01 2.1940E+02 1.0000E+01 1.7177E-01 11 388463 5 65 6

det t wgt psc amfp ddetx radius erg cell nps nch p nrn ipsc
1 4.8092E-04 2.5000E-01 1.5055E+01 1.2837E-01 3.3098E+01 1.0000E+01 1.1808E-01 11 397905 3 61 6

det t wgt psc amfp ddetx radius erg cell nps nch p nrn ipsc
2 6.9698E-05 5.0000E-01 1.3953E+02 1.3767E+00 2.0052E+02 1.0000E+01 1.3000E-01 50 571314 2 42 6

det t wgt psc amfp ddetx radius erg cell nps nch p nrn ipsc
1 5.6973E-04 2.5000E-01 7.3557E+01 1.4935E+00 3.3967E+01 1.0000E+01 9.3916E-02 11 573255 6 88 6

det t wgt psc amfp ddetx radius erg cell nps nch p nrn ipsc
1 2.9775E-03 2.5000E-01 8.8545E+01 1.9940E-01 3.1134E+01 1.0000E+01 1.0968E-01 11 674671 13 254 6

det t wgt psc amfp ddetx radius erg cell nps nch p nrn ipsc
1 3.4525E-03 2.5000E-01 1.1223E+02 1.9960E-01 3.2548E+01 1.0000E+01 1.3000E-01 11 717579 2 48 6

det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
1	6.2456E-04	5.0000E-01	3.2244E+01	1.4263E+00	3.1413E+01	1.0000E+01	1.2246E-01					50	726168	3	45	6
det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
2	1.6632E-04	1.0000E+00	2.2685E+02	1.6681E+00	2.0234E+02	1.0000E+01	1.4318E-01					10	812481	3	60	6
det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
1	5.3368E-04	1.0000E+00	8.8019E+01	2.6691E+00	4.2655E+01	1.0000E+01	1.3000E-01					10	860193	1	13	6
det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
2	8.1664E-05	2.5000E-01	1.1185E+02	2.6078E-01	2.0491E+02	1.0000E+01	1.8570E-01					11	864981	1	21	6
det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
1	1.5431E-03	2.5000E-01	5.5340E+01	2.4316E-01	3.3450E+01	1.0000E+01	1.0500E-01					11	874440	6	101	6
det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
1	5.2532E-04	2.5000E-01	1.5099E+01	4.8062E-02	3.3015E+01	1.0000E+01	1.3000E-01					11	911985	1	15	6
det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
1	4.8100E-04	2.5000E-01	1.5132E+01	1.3574E-01	3.3059E+01	1.0000E+01	1.3000E-01					11	911985	2	23	6
det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
1	1.5455E-03	2.5000E-01	7.6267E+01	5.5582E-01	3.3559E+01	1.0000E+01	9.8397E-02					11	959316	5	95	6
det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
2	6.8770E-05	2.5000E-01	9.9724E+01	2.9810E-01	2.0694E+02	1.0000E+01	1.2357E-01					11	991642	2	37	6
det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
1	1.9707E-03	2.5000E-01	7.4492E+01	1.5406E-01	3.5907E+01	1.0000E+01	1.4937E-01					11	1035243	4	59	6
det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
1	1.1657E-03	2.5000E-01	7.1482E+01	6.2134E-01	3.6204E+01	1.0000E+01	1.4937E-01					11	1035243	5	67	6

det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
1	6.7144E-04	5.0000E-01	6.2578E+01	1.6644E+00	3.7469E+01	1.0000E+01	1.4380E-01	50	1078642	5		86	6			
det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
2	9.4945E-05	2.5000E-01	1.7751E+02	5.5729E-01	2.0641E+02	1.0000E+01	1.8018E-01	11	1087746	3		65	6			
det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
1	8.7006E-04	2.5000E-01	2.9759E+01	1.4314E-01	3.4342E+01	1.0000E+01	1.2527E-01	11	1111829	7		110	6			
det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
1	4.7657E-04	2.5000E-01	2.9136E+01	7.0819E-01	3.4614E+01	1.0000E+01	1.2527E-01	11	1111829	9		129	6			
det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
1	1.4361E-03	2.5000E-01	8.6140E+01	2.7980E-01	4.2475E+01	1.0000E+01	1.2024E-01	11	1169084	4		84	6			
det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
1	5.0758E-04	2.5000E-01	2.4111E+01	2.5089E-01	3.8349E+01	1.0000E+01	1.1081E-01	11	1204350	5		99	6			
det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
1	7.1728E-04	5.0000E-01	9.5580E+01	1.9686E+00	3.8483E+01	1.0000E+01	1.4930E-01	50	1232639	3		59	6			
det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
2	7.5601E-05	5.0000E-01	1.0312E+02	9.9309E-01	2.0052E+02	1.0000E+01	1.3000E-01	50	1318369	1		17	6			
det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
1	5.6998E-04	2.5000E-01	4.5313E+01	9.8667E-01	3.4341E+01	1.0000E+01	1.0458E-01	11	1385745	4		88	6			
det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
2	6.6728E-05	1.0000E+00	9.3301E+01	1.6266E+00	2.0916E+02	1.0000E+01	1.6869E-01	10	1395745	3		70	6			
det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
1	5.5334E-04	2.5000E-01	1.5005E+01	8.8175E-02	3.1430E+01	1.0000E+01	1.3375E-01	11	1430896	3		46	6			
det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			

1 1.0282E-03 5.0000E-01 5.9984E+01 1.4429E+00 3.3116E+01 1.0000E+01 1.3000E-01 50 1572051 6 108 6

det t wgt psc amfp ddetx radius erg cell nps nch p nrn ipsc
2 6.6336E-05 2.5000E-01 8.0864E+01 1.8158E-01 2.0112E+02 1.0000E+01 1.0217E-01 11 1588704 4 85 6

det t wgt psc amfp ddetx radius erg cell nps nch p nrn ipsc
1 5.0668E-04 2.5000E-01 2.5289E+01 5.6608E-01 3.3578E+01 1.0000E+01 1.7126E-01 11 1655950 7 102 6

det t wgt psc amfp ddetx radius erg cell nps nch p nrn ipsc
1 6.3800E-04 2.5000E-01 2.5121E+01 3.3871E-01 3.3414E+01 1.0000E+01 1.7126E-01 11 1655950 8 110 6

det t wgt psc amfp ddetx radius erg cell nps nch p nrn ipsc
1 8.5434E-04 2.5000E-01 2.4413E+01 1.4079E-01 3.1427E+01 1.0000E+01 1.8570E-01 11 1664001 3 39 6

det t wgt psc amfp ddetx radius erg cell nps nch p nrn ipsc
1 4.9346E-04 2.5000E-01 1.9930E+01 1.3594E-01 3.7453E+01 1.0000E+01 1.8570E-01 11 1695063 1 20 6

det t wgt psc amfp ddetx radius erg cell nps nch p nrn ipsc
1 1.1612E-03 5.0000E-01 1.2665E+02 1.8239E+00 3.7428E+01 1.0000E+01 1.4380E-01 50 1771399 6 99 6

det t wgt psc amfp ddetx radius erg cell nps nch p nrn ipsc
1 5.2350E-04 2.5000E-01 7.0521E+01 1.5161E+00 3.4306E+01 1.0000E+01 9.8239E-02 11 1782361 16 292 6

det t wgt psc amfp ddetx radius erg cell nps nch p nrn ipsc
2 9.3573E-05 2.5000E-01 1.0677E+02 5.8288E-02 2.0695E+02 1.0000E+01 1.1075E-01 11 1800281 2 40 6

det t wgt psc amfp ddetx radius erg cell nps nch p nrn ipsc
1 5.0019E-04 2.5000E-01 3.6268E+01 4.2032E-01 4.3531E+01 1.0000E+01 1.2915E-01 11 1811526 13 236 6

det t wgt psc amfp ddetx radius erg cell nps nch p nrn ipsc
1 5.2333E-04 2.5000E-01 2.9283E+01 2.1129E-01 4.2454E+01 1.0000E+01 1.7037E-01 11 1833593 8 136 6

det t wgt psc amfp ddetx radius erg cell nps nch p nrn ipsc
2 1.5870E-04 5.0000E-01 1.6024E+02 6.9714E-01 2.0004E+02 1.0000E+01 1.7501E-01 50 1869062 2 38 6

det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
1	7.5992E-04	2.5000E-01	4.3218E+01	7.8143E-01	3.2184E+01	1.0000E+01	1.0812E-01					11	1925119	4	58	6
det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
1	9.3520E-04	2.5000E-01	3.2528E+01	9.9967E-02	3.5387E+01	1.0000E+01	1.3000E-01					11	1925442	2	26	6
det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
2	1.5937E-04	2.5000E-01	2.2703E+02	3.1759E-01	2.0312E+02	1.0000E+01	1.8570E-01					11	1956285	8	105	6
det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
1	4.7675E-04	5.0000E-01	2.4336E+01	1.2345E+00	3.4381E+01	1.0000E+01	1.8570E-01					50	2159382	3	40	6
det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
1	2.0617E-03	5.0000E-01	9.7782E+01	1.0548E+00	3.6255E+01	1.0000E+01	1.4656E-01					50	2166494	2	40	6
det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
1	5.4699E-04	5.0000E-01	9.1756E+01	2.0320E+00	4.1829E+01	1.0000E+01	1.4091E-01					50	2183252	5	80	6
det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
1	6.1567E-04	2.5000E-01	5.8573E+01	1.0186E+00	3.6971E+01	1.0000E+01	9.2406E-02					11	2319969	7	124	6
det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
1	1.2852E-03	2.5000E-01	1.3956E+02	7.2292E-01	4.5792E+01	1.0000E+01	1.8570E-01					11	2334214	2	44	6
det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
1	1.1976E-03	5.0000E-01	5.3714E+01	1.2483E+00	3.2005E+01	1.0000E+01	1.2771E-01					50	2355712	4	71	6
det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
1	1.1893E-03	5.0000E-01	8.8997E+01	1.4704E+00	3.6995E+01	1.0000E+01	1.2044E-01					50	2443068	2	41	6
det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
1	1.6740E-03	5.0000E-01	1.2254E+02	1.6349E+00	3.3701E+01	1.0000E+01	1.3000E-01					50	2452650	5	92	6

det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
2	6.1736E-05	2.5000E-01	1.1410E+02	5.3949E-01	2.0706E+02	1.0000E+01	1.4751E-01					11	2497606	13	209	6
det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
1	8.9798E-04	2.5000E-01	6.1676E+01	8.2533E-01	3.4601E+01	1.0000E+01	1.3000E-01					11	2556120	2	46	6
det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
1	5.2836E-04	2.5000E-01	1.9075E+01	1.7225E-01	3.4773E+01	1.0000E+01	1.1794E-01					11	2577928	3	55	6
det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
1	3.2627E-03	5.0000E-01	1.9452E+02	9.4308E-01	4.2983E+01	1.0000E+01	1.7960E-01					50	2625505	4	60	6
det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
1	5.3011E-04	2.5000E-01	6.7960E+01	7.7162E-01	4.8559E+01	1.0000E+01	1.4380E-01					11	2647549	7	90	6
det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
2	8.0678E-05	5.0000E-01	1.8015E+02	1.4568E+00	2.0347E+02	1.0000E+01	1.4380E-01					50	2729735	2	25	6
det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
1	5.1116E-04	5.0000E-01	4.2447E+01	1.5137E+00	3.8136E+01	1.0000E+01	1.3000E-01					50	2780623	5	82	6
det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
1	6.2670E-04	2.5000E-01	2.0365E+01	2.5639E-01	3.1631E+01	1.0000E+01	1.3000E-01					11	2878562	2	24	6
det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
1	6.8741E-04	2.5000E-01	9.0621E+01	9.0706E-01	4.6017E+01	1.0000E+01	1.4380E-01					11	2942792	11	207	6
det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
1	3.2928E-03	2.5000E-01	1.1130E+02	2.9941E-01	3.1574E+01	1.0000E+01	1.7372E-01					11	3091738	2	39	6
det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
1	6.2637E-04	2.5000E-01	2.9437E+01	5.2772E-01	3.3214E+01	1.0000E+01	1.3000E-01					11	3116909	6	102	6
det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			

1 5.8600E-04 2.5000E-01 7.7180E+01 1.0494E+00 4.2836E+01 1.0000E+01 1.3000E-01 11 3134216 7 87 6

det t wgt psc amfp ddetx radius erg cell nps nch p nrn ipsc
1 4.8181E-04 2.5000E-01 3.0576E+01 2.9797E-01 4.3294E+01 1.0000E+01 1.3000E-01 11 3161023 1 15 6

det t wgt psc amfp ddetx radius erg cell nps nch p nrn ipsc
1 5.4533E-04 2.5000E-01 4.4832E+01 1.0679E+00 3.3531E+01 1.0000E+01 1.0656E-01 11 3185384 9 161 6

det t wgt psc amfp ddetx radius erg cell nps nch p nrn ipsc
2 6.3824E-05 2.5000E-01 7.9431E+01 1.8690E-01 2.0267E+02 1.0000E+01 1.3000E-01 11 3185455 3 35 6

det t wgt psc amfp ddetx radius erg cell nps nch p nrn ipsc
1 5.9247E-04 2.5000E-01 3.7973E+01 6.9944E-01 3.5596E+01 1.0000E+01 1.2844E-01 11 3195092 2 36 6

det t wgt psc amfp ddetx radius erg cell nps nch p nrn ipsc
1 8.7236E-04 2.5000E-01 5.6743E+01 7.8042E-01 3.4437E+01 1.0000E+01 1.3000E-01 11 3220944 11 193 6

det t wgt psc amfp ddetx radius erg cell nps nch p nrn ipsc
1 5.7987E-04 2.5000E-01 6.0033E+01 1.0150E+00 3.8636E+01 1.0000E+01 1.2286E-01 11 3245009 2 36 6

det t wgt psc amfp ddetx radius erg cell nps nch p nrn ipsc
1 4.6707E-04 5.0000E-01 3.4993E+01 1.3552E+00 3.9211E+01 1.0000E+01 1.6379E-01 50 3276500 2 36 6

det t wgt psc amfp ddetx radius erg cell nps nch p nrn ipsc
1 4.8260E-04 5.0000E-01 3.1520E+01 1.4794E+00 3.4406E+01 1.0000E+01 1.2549E-01 50 3347693 4 71 6

det t wgt psc amfp ddetx radius erg cell nps nch p nrn ipsc
1 5.1747E-04 5.0000E-01 4.9909E+01 1.4486E+00 4.2460E+01 1.0000E+01 1.3669E-01 50 3357499 4 90 6

det t wgt psc amfp ddetx radius erg cell nps nch p nrn ipsc
1 1.2658E-03 5.0000E-01 8.5492E+01 1.4898E+00 3.4807E+01 1.0000E+01 1.3000E-01 50 3361939 1 16 6

det t wgt psc amfp ddetx radius erg cell nps nch p nrn ipsc
1 1.3289E-03 2.5000E-01 8.5079E+01 7.6544E-01 3.4421E+01 1.0000E+01 1.3000E-01 11 3361939 5 57 6

det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
1	1	1.1435E-03	2.5000E-01	7.2440E+01	4.1929E-01	4.0711E+01	1.0000E+01	1.2771E-01				11	3367095	2	34	6
det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
1	1	8.5953E-04	5.0000E-01	8.8525E+01	2.0934E+00	3.1785E+01	1.0000E+01	1.0117E-01				50	3394891	2	41	6
det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
2	1	7.4288E-05	5.0000E-01	9.8099E+01	9.3087E-01	2.0353E+02	1.0000E+01	1.8570E-01				50	3442182	2	39	6
det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
1	1	1.3308E-03	2.5000E-01	1.2372E+02	8.8648E-01	3.9043E+01	1.0000E+01	1.5373E-01				11	3537462	12	232	6
det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
1	5	5.0014E-04	2.5000E-01	3.3212E+01	2.3959E-01	4.5599E+01	1.0000E+01	1.3000E-01				11	3539926	2	27	6
det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
1	5	5.3956E-04	5.0000E-01	2.1833E+01	1.1026E+00	3.2697E+01	1.0000E+01	1.8570E-01				50	3545089	1	14	6
det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
1	5	5.6292E-04	2.5000E-01	1.6586E+01	1.1803E-01	3.2278E+01	1.0000E+01	1.8283E-01				11	3650364	2	35	6
det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
1	2	2.2950E-03	2.5000E-01	9.2796E+01	3.3100E-01	3.3992E+01	1.0000E+01	1.3000E-01				11	3650975	3	33	6
det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
2	9	7.7066E-05	1.0000E+00	2.6143E+02	2.2765E+00	2.0976E+02	1.0000E+01	1.3000E-01				10	3654321	1	16	6
det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
1	1	1.1695E-03	2.5000E-01	5.1011E+01	6.0644E-01	3.0763E+01	1.0000E+01	1.1012E-01				11	3848369	4	73	6
det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
1	1	6.6962E-04	2.5000E-01	7.6065E+01	8.3818E-01	4.4213E+01	1.0000E+01	1.7413E-01				11	3905496	4	71	6

det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
2	9.6599E-05	2.5000E-01	1.1998E+02	1.0080E-01	2.1137E+02	1.0000E+01	1.8479E-01	11	3906556	2		32	6			

det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
2	6.7549E-05	2.5000E-01	9.7349E+01	2.9342E-01	2.0679E+02	1.0000E+01	1.3000E-01	11	3907053	6		109	6			

det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
2	7.7800E-05	2.5000E-01	1.0680E+02	3.0907E-01	2.0024E+02	1.0000E+01	1.0923E-01	11	3921665	7		112	6			

det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
1	8.6059E-04	1.0000E+00	4.1885E+01	1.6285E+00	3.8987E+01	1.0000E+01	1.8570E-01	10	3930467	1		13	6			

det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
1	3.2872E-03	2.5000E-01	2.5142E+02	7.6437E-01	3.7643E+01	1.0000E+01	1.8570E-01	11	4059754	2		41	6			

det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
1	5.4959E-04	5.0000E-01	7.4183E+01	2.1380E+00	3.5585E+01	1.0000E+01	1.2589E-01	50	4079838	3		59	6			

det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
1	9.3644E-04	5.0000E-01	4.5728E+01	1.0991E+00	3.5982E+01	1.0000E+01	1.8570E-01	50	4156195	1		14	6			

det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
1	7.2649E-04	5.0000E-01	4.6224E+01	1.3520E+00	3.6194E+01	1.0000E+01	1.8570E-01	50	4156195	5		69	6			

det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
1	1.3817E-03	2.5000E-01	5.9445E+01	3.0319E-01	3.5554E+01	1.0000E+01	9.3136E-02	11	4159952	6		103	6			

det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
1	2.0539E-03	5.0000E-01	1.1802E+02	1.2470E+00	3.6250E+01	1.0000E+01	1.3000E-01	50	4171884	1		16	6			

det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
1	5.3407E-04	5.0000E-01	4.4289E+01	1.3847E+00	4.0649E+01	1.0000E+01	1.3000E-01	50	4182657	1		14	6			

det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc

1 6.2597E-04 5.0000E-01 5.3306E+01 1.6796E+00 3.5545E+01 1.0000E+01 1.5492E-01 50 4265308 2 35 6

det t wgt psc amfp ddetx radius erg cell nps nch p nrn ipsc
2 1.2845E-04 2.5000E-01 1.3873E+02 4.2095E-02 2.0298E+02 1.0000E+01 1.8570E-01 11 4310093 4 64 6

det t wgt psc amfp ddetx radius erg cell nps nch p nrn ipsc
1 7.0270E-04 2.5000E-01 2.8470E+01 3.4209E-01 3.3838E+01 1.0000E+01 1.3000E-01 11 4328280 2 23 6

det t wgt psc amfp ddetx radius erg cell nps nch p nrn ipsc
1 6.2254E-04 5.0000E-01 7.6286E+01 1.5238E+00 4.6095E+01 1.0000E+01 1.4380E-01 50 4354629 1 14 6

det t wgt psc amfp ddetx radius erg cell nps nch p nrn ipsc
1 5.1419E-04 5.0000E-01 1.9935E+01 9.4154E-01 3.4688E+01 1.0000E+01 2.0530E-01 50 4398274 1 14 6

det t wgt psc amfp ddetx radius erg cell nps nch p nrn ipsc
1 1.3410E-03 1.0000E+00 2.7016E+02 2.7793E+00 4.4615E+01 1.0000E+01 1.2317E-01 10 4535430 2 36 6

det t wgt psc amfp ddetx radius erg cell nps nch p nrn ipsc
1 1.9512E-03 5.0000E-01 1.2248E+02 1.6198E+00 3.1444E+01 1.0000E+01 1.2842E-01 50 4573242 9 197 6

det t wgt psc amfp ddetx radius erg cell nps nch p nrn ipsc
2 1.0099E-04 5.0000E-01 1.9850E+02 1.2931E+00 2.0718E+02 1.0000E+01 1.8570E-01 50 4594708 2 24 6

det t wgt psc amfp ddetx radius erg cell nps nch p nrn ipsc
1 6.2792E-04 5.0000E-01 4.3828E+01 1.4149E+00 3.6735E+01 1.0000E+01 1.1670E-01 50 4608539 2 41 6

det t wgt psc amfp ddetx radius erg cell nps nch p nrn ipsc
1 4.7662E-04 1.0000E+00 1.4555E+02 3.4950E+00 3.8406E+01 1.0000E+01 1.0012E-01 10 4613891 3 62 6

det t wgt psc amfp ddetx radius erg cell nps nch p nrn ipsc
1 6.7040E-04 5.0000E-01 4.4527E+01 1.4417E+00 3.5358E+01 1.0000E+01 1.3000E-01 50 4681007 5 52 6

det t wgt psc amfp ddetx radius erg cell nps nch p nrn ipsc
2 7.3379E-05 1.0000E+00 7.0702E+01 1.2945E+00 2.0499E+02 1.0000E+01 1.8570E-01 10 4684360 1 16 6

det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
1	1.0558E-03	2.5000E-01	4.6122E+01	2.6416E-01	3.6532E+01	1.0000E+01	1.3000E-01					11	4703340	2	32	6
det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
1	1.5123E-03	2.5000E-01	8.6092E+01	8.4980E-01	3.1118E+01	1.0000E+01	1.3000E-01					11	4712733	2	23	6
det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
1	1.7255E-03	2.5000E-01	8.5665E+01	7.1743E-01	3.1048E+01	1.0000E+01	1.3000E-01					11	4712733	7	97	6
det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
1	2.9430E-03	2.5000E-01	1.1299E+02	2.5841E-01	3.4348E+01	1.0000E+01	1.4855E-01					11	4764056	8	137	6
det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
1	6.1087E-04	2.5000E-01	2.3456E+01	3.8734E-02	3.8338E+01	1.0000E+01	1.1820E-01					11	4768139	6	100	6
det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
2	8.9505E-05	2.5000E-01	9.3860E+01	3.4463E-02	2.0078E+02	1.0000E+01	1.8570E-01					11	4879138	1	15	6
det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
2	9.3422E-05	2.5000E-01	1.7357E+02	3.4902E-01	2.2835E+02	1.0000E+01	1.6992E-01					11	4985150	7	106	6
det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
2	7.9443E-05	2.5000E-01	1.3739E+02	5.1180E-01	2.0309E+02	1.0000E+01	1.2469E-01					11	5166101	16	311	6
det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
2	6.3427E-05	5.0000E-01	1.4501E+02	1.3501E+00	2.1716E+02	1.0000E+01	1.8570E-01					50	5240953	1	23	6
det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
2	9.2058E-05	2.5000E-01	1.2053E+02	2.2794E-01	2.0366E+02	1.0000E+01	1.8570E-01					11	5309294	1	17	6
det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
2	8.1922E-05	2.5000E-01	1.0587E+02	2.1140E-01	2.0402E+02	1.0000E+01	1.3000E-01					11	5507091	5	73	6

det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc				
2	7.1386E-05	2.5000E-01	1.6026E+02	6.6252E-01	2.1460E+02	1.0000E+01	1.6330E-01					11	5607881	4		52	6

det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc				
2	1.2320E-04	5.0000E-01	1.5423E+02	8.2608E-01	2.0883E+02	1.0000E+01	1.6773E-01					50	5742616	4		95	6

det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc				
2	6.7748E-05	2.5000E-01	1.4851E+02	6.1173E-01	2.1751E+02	1.0000E+01	1.6067E-01					11	5798661	6		98	6

det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc				
2	1.4616E-04	5.0000E-01	2.5270E+02	1.1450E+00	2.0924E+02	1.0000E+01	1.8570E-01					50	5941969	1		15	6

det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc				
2	6.3250E-05	5.0000E-01	1.0746E+02	1.1696E+00	2.0489E+02	1.0000E+01	1.5957E-01					50	6106548	3		63	6

det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc				
2	7.7663E-05	2.5000E-01	1.1530E+02	3.6248E-01	2.0275E+02	1.0000E+01	1.3000E-01					11	6122751	3		40	6

det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc				
2	8.2003E-05	5.0000E-01	8.5367E+01	6.7714E-01	2.0516E+02	1.0000E+01	1.7525E-01					50	6128781	6		107	6

det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc				
2	8.0096E-05	2.5000E-01	8.1642E+01	1.4839E-02	1.9990E+02	1.0000E+01	1.3000E-01					11	6301952	1		16	6

det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc				
2	7.7647E-05	2.5000E-01	1.1386E+02	2.1132E-01	2.1732E+02	1.0000E+01	1.6915E-01					11	6325708	2		44	6

det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc				
2	8.0105E-05	2.5000E-01	9.7521E+01	9.2966E-02	2.1009E+02	1.0000E+01	1.2952E-01					11	6569431	8		143	6

det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc				
2	6.3612E-05	5.0000E-01	9.4114E+01	1.0530E+00	2.0267E+02	1.0000E+01	1.3000E-01					50	6864922	1		15	6

det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc

det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
2	8.5899E-05	2.5000E-01	9.9501E+01	5.7195E-02	2.0863E+02	1.0000E+01	1.3000E-01	11	6957752	3	47	6				
det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
2	8.3686E-05	2.5000E-01	1.8246E+02	6.7611E-01	2.1005E+02	1.0000E+01	1.8570E-01	11	6982788	1	22	6				
det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
2	1.4716E-04	5.0000E-01	1.8785E+02	8.4998E-01	2.0837E+02	1.0000E+01	1.4865E-01	50	7006398	5	97	6				
det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
2	8.0744E-05	1.0000E+00	2.2806E+02	2.3206E+00	2.1012E+02	1.0000E+01	1.3000E-01	10	7015682	1	14	6				
det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
2	7.7876E-05	2.5000E-01	9.8330E+01	2.0527E-01	2.0228E+02	1.0000E+01	1.0346E-01	11	7071106	7	116	6				
det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
2	7.8800E-05	2.5000E-01	1.1116E+02	2.6310E-01	2.0772E+02	1.0000E+01	1.4380E-01	11	7237656	3	44	6				
det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
2	8.2322E-05	2.5000E-01	9.5347E+01	6.8997E-02	2.0739E+02	1.0000E+01	1.0388E-01	11	7430193	8	150	6				
det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
2	7.0185E-05	5.0000E-01	1.4111E+02	1.3704E+00	2.0159E+02	1.0000E+01	1.2327E-01	50	7480406	3	46	6				
det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
2	1.9834E-04	5.0000E-01	2.8574E+02	9.6830E-01	2.0865E+02	1.0000E+01	2.0210E-01	50	7980777	6	106	6				
det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
2	8.0021E-05	2.5000E-01	1.0923E+02	2.7107E-01	2.0351E+02	1.0000E+01	1.3000E-01	11	7983577	6	88	6				
det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
2	1.0526E-04	2.5000E-01	1.3134E+02	4.5678E-02	2.1778E+02	1.0000E+01	1.2624E-01	11	8224664	6	117	6				
det	t	wgt	psc	amfp	ddetx	radius	erg	cell	nps	nch	p	nrn	ipsc			
2	8.7746E-05	2.5000E-01	1.0136E+02	7.5749E-02	2.0642E+02	1.0000E+01	1.8570E-01	11	8660330	2	35	6				

```
det   t      wgt      psc      amfp      ddetx     radius     erg      cell      nps  nch  p          nrn    ipsc
 2 8.7664E-05 2.5000E-01 1.1636E+02 7.3317E-02 2.2154E+02 1.0000E+01 1.6760E-01   11    8932238  8          138     6

det   t      wgt      psc      amfp      ddetx     radius     erg      cell      nps  nch  p          nrn    ipsc
 2 7.1299E-05 5.0000E-01 1.2433E+02 1.2120E+00 2.0321E+02 1.0000E+01 1.2898E-01   50    9142262  5          67      6

det   t      wgt      psc      amfp      ddetx     radius     erg      cell      nps  nch  p          nrn    ipsc
 2 6.6424E-05 2.5000E-01 9.6398E+01 3.4061E-01 2.0267E+02 1.0000E+01 1.0685E-01   11    9325623  14         248     6

det   t      wgt      psc      amfp      ddetx     radius     erg      cell      nps  nch  p          nrn    ipsc
 2 7.6836E-05 2.5000E-01 9.5882E+01 2.1270E-01 2.0034E+02 1.0000E+01 1.0768E-01   11    9602391  10         167     6

det   t      wgt      psc      amfp      ddetx     radius     erg      cell      nps  nch  p          nrn    ipsc
 2 1.0410E-04 2.5000E-01 1.4581E+02 2.4234E-01 2.0913E+02 1.0000E+01 1.4970E-01   11    9649833  4          82      6

det   t      wgt      psc      amfp      ddetx     radius     erg      cell      nps  nch  p          nrn    ipsc
 2 7.4218E-05 2.5000E-01 8.0648E+01 7.2804E-02 2.0050E+02 1.0000E+01 1.4349E-01   11    9681295  5          94      6

det   t      wgt      psc      amfp      ddetx     radius     erg      cell      nps  nch  p          nrn    ipsc
 2 6.6683E-05 5.0000E-01 1.1436E+02 1.1936E+00 2.0340E+02 1.0000E+01 1.3000E-01   50    9952165  1          16      6
1problem summary

    run terminated when    10000000  particle histories were done.
+                                                                    12/15/15 16:50:13
    Cimmarron Resin Bed Dose Estimation                              probid =  12/15/15 16:46:08

photon creation    tracks     weight     energy          photon loss      tracks     weight     energy
                          (per source particle)                                (per source particle)

source          10000000   1.0000E+00   1.0453E-01      escape          1609553   4.0239E-02   5.5397E-03
nucl. interaction     0   0.           0.              energy cutoff         0   0.           4.5751E-06
particle decay        0   0.           0.              time cutoff           0   0.           0.
weight window         0   0.           0.              weight window         0   0.           0.
```

```
cell importance    8402371    2.0823E-02    1.8945E-03        cell importance    557774    2.0827E-02    1.8943E-03
weight cutoff          0    0.            0.                  weight cutoff          0    0.            0.
e or t importance      0    0.            0.                  e or t importance      0    0.            0.
dxtran                 0    0.            0.                  dxtran                 0    0.            0.
forced collisions      0    0.            0.                  forced collisions      0    0.            0.
exp. transform         0    0.            0.                  exp. transform         0    0.            0.
from neutrons          0    0.            0.                  compton scatter        0    0.            5.0425E-02
bremsstrahlung     289865    1.2283E-02    1.1847E-04        capture            20289522    1.1232E+00    4.9640E-02
p-annihilation         0    0.            0.                  pair production        0    0.            0.
photonuclear           0    0.            0.                  photonuclear abs       0    0.            0.
electron x-rays        0    0.            0.                  loss to photofis       0    0.            0.
compton fluores        0    0.            0.
muon capt fluores      0    0.            0.
1st fluorescence   3764613    1.5121E-01    9.6138E-04
2nd fluorescence       0    0.            0.
(gamma,xgamma)         0    0.            0.
tabular sampling       0    0.            0.
prompt photofis        0    0.            0.
   total       22456849    1.1843E+00    1.0750E-01           total       22456849    1.1843E+00    1.0750E-01

  number of photons banked              7207156      average time of (shakes)          cutoffs
  photon tracks per source particle     2.2457E+00     escape          2.1082E-01       tco  1.0000E+33
  photon collisions per source particle 7.3644E+00     capture         1.5924E-01       eco  1.0000E-03
  total photon collisions              73644179      capture or escape 1.6103E-01       wc1 -5.0000E-01
                                                     any termination   1.6002E-01       wc2 -2.5000E-01

computer time so far in this run  236.73 minutes        maximum number ever in bank      5
computer time in mcrun            236.60 minutes        bank overflows to backup file    0
source particles per minute         4.2266E+04
random numbers generated          1463531465          most random numbers used was      1114 in history    3631177

range of sampled source weights = 1.0000E+00 to 1.0000E+00

source efficiency = 1.0000  in cell      10
```

number of histories processed by each thread
1312074 1201267 1259672 1299659 1294866 1282904 1264634 1084924
1photon activity in each cell print table 126

cell	tracks entering	population	collisions * weight (per history)	collisions weighted energy	number weighted energy	flux average (relative)	average track weight	average track mfp (cm)
1	10	10322059	10193584	50738870	5.0739E+00	8.3033E-02	8.3033E-02	1.0000E+00	1.1545E+01
2	50	9023654	8922895	13144162	6.5721E-01	1.1558E-01	1.1558E-01	1.0000E+00	4.3220E-01
3	11	6437568	6528077	9761147	2.4403E-01	1.2524E-01	1.2524E-01	1.0000E+00	4.8793E-01
4	30	1376517	1376517	0	0.0000E+00	1.3735E-01	1.3735E-01	2.0000E+00	0.0000E+00
total		27159798	27021073	73644179	5.9751E+00			

1tally 15 nps = 10000000
tally type 5 particle flux at a point detector.
particle(s): photons
this tally is modified by standard dose function 1.

this tally is all multiplied by 1.50451E+08

detector located at x,y,z = 9.14300E+01 0.00000E+00 6.91975E+01
2.39237E-05 0.0055

detector located at x,y,z = 9.14300E+01 0.00000E+00 6.91975E+01
uncollided photon flux
7.28378E-06 0.0018

detector score diagnostics times average score	cumulative fraction of transmissions	tally per transmissions	cumulative fraction of history	total tally
1.00000E-01	2066474	0.33337	9.19114E-09	0.02066

1.00000E+00	2782476	0.78225	4.22849E-08	0.11571
2.00000E+00	480224	0.85973	3.02936E-08	0.18381
5.00000E+00	443038	0.93120	6.23896E-08	0.32405
1.00000E+01	215883	0.96603	6.75746E-08	0.47595
1.00000E+02	202626	0.99871	1.84876E-07	0.89153
1.00000E+03	2350	0.99909	2.79066E-08	0.95426
1.00000E+38	195	0.99913	2.03054E-08	0.99991
before dd roulette	5422	1.00000	4.07592E-11	1.00000

average tally per history = 4.44862E-07 largest score = 5.32061E-03
(largest score)/(average tally) = 1.19601E+04 nps of largest score = 313718

score contributions by cell

cell		misses	hits	tally per history	weight per hit
1	10	50103740	5496658	2.24457E-07	4.08352E-07
2	50	6049166	355174	8.06515E-08	2.27076E-06
3	11	5057541	346856	1.39754E-07	4.02916E-06
	total	61210447	6198688	4.44862E-07	7.17672E-07

score misses

russian roulette on pd	0
psc=0.	975368
russian roulette in transmission	56695829
underflow in transmission	3539250
hit a zero-importance cell	0
energy cutoff	0

==========

results of 10 statistical checks for the estimated answer for the tally fluctuation chart (tfc) bin of tally 15

tfc bin behavior	--mean-- behavior	---------relative error--------- value	decrease	decrease rate	----variance of the variance---- value	decrease	decrease rate	--figure of merit-- value	-pdf- behavior	slope
desired	random	<0.05	yes	1/sqrt(nps)	<0.10	yes	1/nps	constant	random	>3.00
observed	random	0.01	yes	yes	0.02	yes	yes	constant	random	3.12
passed?	yes	yes	yes	yes	yes	yes	yes	yes	yes	yes

==========

this tally meets the statistical criteria used to form confidence intervals: check the tally fluctuation chart to verify.
the results in other bins associated with this tally may not meet these statistical criteria.

----- estimated confidence intervals: -----

estimated asymmetric confidence interval(1,2,3 sigma): 2.3800E-05 to 2.4061E-05; 2.3669E-05 to 2.4192E-05; 2.3538E-05 to 2.4323E-05
estimated symmetric confidence interval(1,2,3 sigma): 2.3793E-05 to 2.4055E-05; 2.3662E-05 to 2.4185E-05; 2.3531E-05 to 2.4316E-05

1analysis of the results in the tally fluctuation chart bin (tfc) for tally 15 with nps = 10000000 print table 160

normed average tally per history = 2.39237E-05 unnormed average tally per history = 2.39237E-05
estimated tally relative error = 0.0055 estimated variance of the variance = 0.0190
relative error from zero tallies = 0.0005 relative error from nonzero scores = 0.0055

number of nonzero history tallies = 3040522 efficiency for the nonzero tallies = 0.3041
history number of largest tally = 313718 largest unnormalized history tally = 3.74298E-01
(largest tally)/(average tally) = 1.56455E+04 (largest tally)/(avg nonzero tally)= 4.75705E+03

(confidence interval shift)/mean = 0.0003 shifted confidence interval center = 2.39305E-05

if the largest history score sampled so far were to occur on the next history, the tfc bin quantities would change as follows:

estimated quantities	value at nps	value at nps+1	value(nps+1)/value(nps)-1.
mean	2.39237E-05	2.39611E-05	0.001564
relative error	5.47129E-03	5.68168E-03	0.038452
variance of the variance	1.90457E-02	2.19879E-02	0.154486
shifted center	2.39305E-05	2.39314E-05	0.000038
figure of merit	1.41191E+02	1.30928E+02	-0.072686

the estimated inverse power slope of the 200 largest tallies starting at 2.58402E-02 is 3.1203
the history score probability density function appears to have an unsampled region at the largest history scores:
please examine. see print table 161.

fom = (histories/minute)*(f(x) signal-to-noise ratio)**2 = (4.227E+04)*(5.780E-02)**2 = (4.227E+04)*(3.341E-03) = 1.412E+02

1tally 25 nps = 10000000
tally type 5 particle flux at a point detector.
particle(s): photons
this tally is modified by standard dose function 1.

this tally is all multiplied by 2.25676E+07

detector located at x,y,z = 9.14300E+01 2.43840E+02 6.91975E+01
4.97439E-07 0.0024

detector located at x,y,z = 9.14300E+01 2.43840E+02 6.91975E+01
uncollided photon flux
1.65726E-07 0.0010

detector score diagnostics		cumulative	tally	cumulative
		fraction of	per	fraction of
times average score	transmissions	transmissions	history	total tally
1.00000E-01	2719757	0.28347	1.67735E-09	0.02719

1.00000E+00	4570832	0.75986	1.01499E-08	0.19169
2.00000E+00	976550	0.86165	8.59433E-09	0.33098
5.00000E+00	959927	0.96169	1.86191E-08	0.63274
1.00000E+01	304493	0.99343	1.27090E-08	0.83871
1.00000E+02	55103	0.99917	6.01146E-09	0.93614
1.00000E+03	2382	0.99942	3.38705E-09	0.99104
1.00000E+38	61	0.99943	5.47142E-10	0.99990
before dd roulette	5491	1.00000	5.86511E-12	1.00000

average tally per history = 6.17012E-08 largest score = 1.98339E-04
(largest score)/(average tally) = 3.21450E+03 nps of largest score = 7980777

score contributions by cell

cell	misses	hits	tally per history	weight per hit
1 10	47440116	8160282	3.45288E-08	4.23132E-08
2 50	5631580	772760	1.10836E-08	1.43429E-07
3 11	4742843	661554	1.60889E-08	2.43198E-07
total	57814539	9594596	6.17012E-08	6.43083E-08

score misses

russian roulette on pd	0
psc=0.	855422
russian roulette in transmission	53023942
underflow in transmission	3935175
hit a zero-importance cell	0
energy cutoff	0

==========

results of 10 statistical checks for the estimated answer for the tally fluctuation chart (tfc) bin of tally 25

tfc bin behavior	--mean-- behavior	---------relative error--------- value	decrease	decrease rate	----variance of the variance---- value	decrease	decrease rate	--figure of merit-- value	-pdf- behavior	slope
desired	random	<0.05	yes	1/sqrt(nps)	<0.10	yes	1/nps	constant	random	>3.00
observed	random	0.00	yes	yes	0.00	yes	yes	constant	random	4.00
passed?	yes	yes	yes	yes	yes	yes	yes	yes	yes	yes

==========

this tally meets the statistical criteria used to form confidence intervals: check the tally fluctuation chart to verify.
the results in other bins associated with this tally may not meet these statistical criteria.

----- estimated confidence intervals: -----

estimated asymmetric confidence interval(1,2,3 sigma): 4.9627E-07 to 4.9866E-07; 4.9508E-07 to 4.9986E-07; 4.9388E-07 to 5.0105E-07
estimated symmetric confidence interval(1,2,3 sigma): 4.9625E-07 to 4.9863E-07; 4.9505E-07 to 4.9983E-07; 4.9386E-07 to 5.0102E-07

1analysis of the results in the tally fluctuation chart bin (tfc) for tally 25 with nps = 10000000 print table 160

normed average tally per history = 4.97439E-07 unnormed average tally per history = 4.97439E-07
estimated tally relative error = 0.0024 estimated variance of the variance = 0.0047
relative error from zero tallies = 0.0004 relative error from nonzero scores = 0.0024

number of nonzero history tallies = 4079202 efficiency for the nonzero tallies = 0.4079
history number of largest tally = 7980777 largest unnormalized history tally = 2.28189E-03
(largest tally)/(average tally) = 4.58726E+03 (largest tally)/(avg nonzero tally)= 1.87124E+03

(confidence interval shift)/mean = 0.0001 shifted confidence interval center = 4.97467E-07

if the largest history score sampled so far were to occur on the next history, the tfc bin quantities would change as follows:

estimated quantities	value at nps	value at nps+1	value(nps+1)/value(nps)-1.
mean	4.97439E-07	4.97668E-07	0.000459
relative error	2.40083E-03	2.44312E-03	0.017616
variance of the variance	4.71546E-03	5.62879E-03	0.193689
shifted center	4.97467E-07	4.97470E-07	0.000006
figure of merit	7.33269E+02	7.08102E+02	-0.034322

the estimated inverse power slope of the 200 largest tallies starting at 2.97193E-04 is 4.0044
the large score tail of the empirical history score probability density function appears to have no unsampled regions.

fom = (histories/minute)*(f(x) signal-to-noise ratio)**2 = (4.227E+04)*(1.317E-01)**2 = (4.227E+04)*(1.735E-02) = 7.333E+02

1status of the statistical checks used to form confidence intervals for the mean for each tally bin

tally result of statistical checks for the tfc bin (the first check not passed is listed) and error magnitude check for all bins

15 passed the 10 statistical checks for the tally fluctuation chart bin result
passed all bin error check: 2 tally bins all have relative errors less than 0.05 with no zero bins

25 passed the 10 statistical checks for the tally fluctuation chart bin result
passed all bin error check: 2 tally bins all have relative errors less than 0.05 with no zero bins

the 10 statistical checks are only for the tally fluctuation chart bin and do not apply to other tally bins.

1tally fluctuation charts

	tally 15					tally 25				
nps	mean	error	vov	slope	fom	mean	error	vov	slope	fom
512000	2.4094E-05	0.0352	0.5579	2.2	83	4.9978E-07	0.0102	0.1024	3.3	988

1024000	2.4078E-05	0.0213	0.2766	2.2	110	4.9771E-07	0.0075	0.0438	3.2	876
1536000	2.3929E-05	0.0160	0.1849	2.3	136	4.9688E-07	0.0059	0.0257	3.1	989
2048000	2.3935E-05	0.0128	0.1420	2.3	153	4.9847E-07	0.0056	0.0281	3.2	796
2560000	2.3867E-05	0.0111	0.1081	2.4	150	4.9707E-07	0.0049	0.0214	3.8	776
3072000	2.3783E-05	0.0100	0.0890	2.6	160	4.9589E-07	0.0044	0.0171	3.9	832
3584000	2.3846E-05	0.0094	0.0676	3.0	161	4.9585E-07	0.0040	0.0136	4.0	880
4096000	2.3883E-05	0.0088	0.0556	3.0	161	4.9676E-07	0.0037	0.0110	3.5	899
4608000	2.3874E-05	0.0082	0.0469	2.9	161	4.9657E-07	0.0035	0.0098	3.5	870
5120000	2.4010E-05	0.0081	0.0377	2.8	153	4.9693E-07	0.0034	0.0087	3.6	879
5632000	2.3975E-05	0.0076	0.0338	2.6	161	4.9764E-07	0.0032	0.0075	3.9	888
6144000	2.3969E-05	0.0072	0.0300	2.7	157	4.9766E-07	0.0031	0.0069	3.4	824
6656000	2.3965E-05	0.0068	0.0273	2.7	150	4.9714E-07	0.0030	0.0063	3.1	784
7168000	2.3940E-05	0.0065	0.0253	2.7	154	4.9817E-07	0.0029	0.0056	3.3	770
7680000	2.3942E-05	0.0063	0.0236	2.6	150	4.9818E-07	0.0028	0.0051	3.5	777
8192000	2.3982E-05	0.0061	0.0208	3.0	143	4.9807E-07	0.0027	0.0062	3.2	724
8704000	2.3969E-05	0.0059	0.0190	3.1	143	4.9799E-07	0.0026	0.0057	3.2	726
9216000	2.3925E-05	0.0056	0.0184	3.2	144	4.9759E-07	0.0025	0.0053	3.3	721
9728000	2.3939E-05	0.0056	0.0194	3.1	138	4.9753E-07	0.0024	0.0049	3.7	723
10000000	2.3924E-05	0.0055	0.0190	3.1	141	4.9744E-07	0.0024	0.0047	4.0	733

****

dump no. 2 on file runtpe nps = 10000000 coll = 73644179 ctm = 236.60 nrn = 1463531465

5 warning messages so far.

run terminated when 10000000 particle histories were done.

computer time = 236.73 minutes

mcnp version 6 05/08/13 12/15/15 16:50:13 probid = 12/15/15 16:46:08

ENERCON Excellence—Every project. Every day.	CALCULATION PREPARATION CHECKLIST	CALC NO. EPM017-CALC-001 REV. 0 PAGE NO. 1 of 6

CHECKLIST ITEMS[1]	YES	NO	N/A
GENERAL REQUIREMENTS			
1. If the calculation is being performed to a client procedure, is the procedure being used the latest revision? Client procedure is not used in this calculation. ENERCON QA procedures used throughout this project.	☐	☐	☒
2. Are the proper forms being used and are they the latest revision?	☒	☐	☐
3. Have the appropriate client review forms/checklists been completed? Client procedure is not used in this calculation. ENERCON QA procedures used throughout this project.	☐	☐	☒
4. Are all pages properly identified with a calculation number, calculation revision and page number consistent with the requirements of the client's procedure? Client procedure is not used in this calculation. ENERCON QA procedures used throughout this project.	☐	☐	☒
5. Is all information legible and reproducible?	☒	☐	☐
6. Is the calculation presented in a logical and orderly manner?	☒	☐	☐
7. Is there an existing calculation that should be revised or voided?	☐	☒	☐
8. Is it possible to alter an existing calculation instead of preparing a new calculation for this situation? No current ENERCON calculations exist that are similar to this calculation.	☐	☒	☐
9. If an existing calculation is being used for design inputs, are the key design inputs, assumptions and engineering judgments used in that calculation valid and do they apply to the calculation revision being performed.	☒	☐	☐
10. Is the format of the calculation consistent with applicable procedures and expectations?	☒	☐	☐
11. Were design input/output documents properly updated to reference this calculation? No ENERCON design inputs or outputs are affected by this calculation.	☐	☐	☒

CHECKLIST ITEMS[1]	YES	NO	N/A
12. Can the calculation logic, methodology and presentation be properly understood without referring back to the originator for clarification?	☒	☐	☐
OBJECTIVE AND SCOPE			
13. Does the calculation provide a clear concise statement of the problem and objective of the calculation?	☒	☐	☐
14. Does the calculation provide a clear statement of quality classification?	☒	☐	☐
15. Is the reason for performing and the end use of the calculation understood?	☒	☐	☐
16. Does the calculation provide the basis for information found in the plant's license basis? This calculation applies to a remediation site. No work performed in this calculation is applicable to a licensing basis.	☐	☐	☒
17. If so, is this documented in the calculation?	☐	☐	☒
18. Does the calculation provide the basis for information found in the plant's design basis documentation?	☐	☐	☒
19. If so, is this documented in the calculation?	☐	☐	☒
20. Does the calculation otherwise support information found in the plant's design basis documentation?	☐	☐	☒
21. If so, is this documented in the calculation?	☐	☐	☒
22. Has the appropriate design or license basis documentation been revised, or has the change notice or change request documents being prepared for submittal?	☐	☐	☒
DESIGN INPUTS			
23. Are design inputs clearly identified?	☒	☐	☐
24. Are design inputs retrievable or have they been added as attachments?	☒	☐	☐

CHECKLIST ITEMS[1]	YES	NO	N/A
25. If Attachments are used as design inputs or assumptions are the Attachments traceable and verifiable?	☒	☐	☐
26. Are design inputs clearly distinguished from assumptions?	☒	☐	☐
DESIGN INPUTS (Continued)			
27. Does the calculation rely on Attachments for design inputs or assumptions? If yes, are the attachments properly referenced in the calculation?	☒	☐	☐
28. Are input sources (including industry codes and standards) appropriately selected and are they consistent with the quality classification and objective of the calculation?	☒	☐	☐
29. Are input sources (including industry codes and standards) consistent with the plant's design and license basis?	☐	☐	☒
30. If applicable, do design inputs adequately address actual plant conditions?	☐	☐	☒
31. Are input values reasonable and correctly applied?	☒	☐	☐
32. Are design input sources approved? The Cimarron design is currently at 60% completion.	☐	☒	☐
33. Does the calculation reference the latest revision of the design input source?	☒	☐	☐
34. Were all applicable plant operating modes considered?	☐	☐	☒
ASSUMPTIONS			
35. Are assumptions reasonable/appropriate to the objective?	☒	☐	☐
36. Is adequate justification/basis for all assumptions provided?	☒	☐	☐
37. Are any engineering judgments used?	☒	☐	☐
38. Are engineering judgments clearly identified as such? No engineering judgments were applied in this evaluation.	☐	☐	☒

CHECKLIST ITEMS[1]	YES	NO	N/A
39. If engineering judgments are utilized as design inputs, are they reasonable and can they be quantified or substantiated by reference to site or industry standards, engineering principles, physical laws or other appropriate criteria?	☐	☐	☒
METHODOLOGY			
40. Is the methodology used in the calculation described or implied in the plant's licensing basis?	☐	☐	☒
41. If the methodology used differs from that described in the plant's licensing basis, has the appropriate license document change notice been initiated?	☐	☐	☒
42. Is the methodology used consistent with the stated objective?	☒	☐	☐
43. Is the methodology used appropriate when considering the quality classification of the calculation and intended use of the results?	☒	☐	☐
BODY OF CALCULATION			
44. Are equations used in the calculation consistent with recognized engineering practice and the plant's design and license basis?	☒	☐	☐
45. Is there reasonable justification provided for the use of equations not in common use? Equations applied in this evaluation are in common use in the industry.	☐	☐	☒
46. Are the mathematical operations performed properly and documented in a logical fashion?	☒	☐	☐
47. Is the math performed correctly?	☒	☐	☐
48. Have adjustment factors, uncertainties and empirical correlations used in the analysis been correctly applied?	☒	☐	☐
49. Has proper consideration been given to results that may be overly sensitive to very small changes in input? Results generated by calculations performed in this evaluation are not significantly affected by minor perturbations of variables.	☐	☐	☒
SOFTWARE/COMPUTER CODES			
50. Are computer codes or software languages used in the preparation of the calculation?	☒	☐	☐

CHECKLIST ITEMS[1]	YES	NO	N/A
51. Have the requirements of CSP 3.09 for use of computer codes or software languages, including verification of accuracy and applicability been met?	☒	☐	☐
SOFTWARE/COMPUTER CODES (Continued)			
52. Are the codes properly identified along with source vendor, organization, and revision level?	☒	☐	☐
53. Is the computer code applicable for the analysis being performed?	☒	☐	☐
54. If applicable, does the computer model adequately consider actual plant conditions?	☒	☐	☐
55. Are the inputs to the computer code clearly identified and consistent with the inputs and assumptions documented in the calculation?	☒	☐	☐
56. Is the computer output clearly identified?	☒	☐	☐
57. Does the computer output clearly identify the appropriate units? The output units are not identified in the output document. Tallies have been modified through multipliers and dose response functions. This process has been adequately documented within this calculation.	☐	☒	☐
58. Are the computer outputs reasonable when compared to the inputs and what was expected? Only basic functions and operations in Microsoft Excel 2013 were applied in this calculation.	☒	☐	☐
59. Was the computer output reviewed for ERROR or WARNING messages that could invalidate the results? While warning messages exist, they do not impact the results.	☒	☐	☒
RESULTS AND CONCLUSIONS			
60. Is adequate acceptance criteria specified? No acceptance criteria required for this evaluation.	☐	☐	☒
61. Are the stated acceptance criteria consistent with the purpose of the calculation, and intended use?	☐	☐	☒

CHECKLIST ITEMS[1]	YES	NO	N/A
62. Are the stated acceptance criteria consistent with the plant's design basis, applicable licensing commitments and industry codes, and standards?	☐	☐	☒
63. Do the calculation results and conclusions meet the stated acceptance criteria?	☐	☐	☒
64. Are the results represented in the proper units with an appropriate tolerance, if applicable?	☒	☐	☐
65. Are the calculation results and conclusions reasonable when considered against the stated inputs and objectives?	☒	☐	☐
66. Is sufficient conservatism applied to the outputs and conclusions?	☒	☐	☐
67. Do the calculation results and conclusions affect any other calculations? No ENERCON calculations are affected by this evaluation.	☐	☐	☒
68. If so, have the affected calculations been revised?	☐	☐	☒
69. Does the calculation contain any conceptual, unconfirmed or open assumptions requiring later confirmation?	☐	☒	☐
70. If so, are they properly identified?	☐	☐	☒
DESIGN REVIEW			
71. Have alternate calculation methods been used to verify calculation results?	☒	☐	☐

Note:

1. Where required, provide clarification/justification for answers to the questions in the space provided below each question. An explanation is required for any questions answered as "No' or "N/A".

Originator:

Caleb Trainor

12/21/2015

Print Name and Sign

Date